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Rev. 5, Draft A
Redline/Strikeout

Removal Action Work Plan for 105-DR and 105-F Building Interim Safe Storage Projects and Ancillary Buildings

United States
Department of Energy

FOR EXTERNAL REVIEW

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Removal Action Work Plan for 105-DR and 105-F Building Interim Safe Storage Projects and Ancillary Buildings

June 2002

United States Department of Energy

P.O. Box 550, Richland, Washington 99352

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ACRONYMS

ACM	asbestos-containing material
AOC	area of contamination
ALARA	as low as reasonably achievable
ARAR	applicable or relevant and appropriate requirement
BHI	Bechtel Hanford, Inc.
CERCLA	*Comprehensive Environmental Response, Compensation, and Liability Act of 1980*
CFR	*Code of Federal Regulations*
CWC	Central Waste Complex
D&D	decontamination and decommissioning
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
DQO	Data Quality Objectives
Ecology	Washington State Department of Ecology
EDPI	Engineering Department Project Instruction
EE/CA	engineering evaluation/cost analysis
EPA	U.S. Environmental Protection Agency
ERC	Environmental Restoration Contractor
ERDF	Environmental Restoration Disposal Facility
ETF	Effluent Treatment Facility
FEIS	final environmental impact statement
FSB	fuel storage basin
HASP	health and safety plan
HCR	horizontal control rod
HEPA	high efficiency particulate air
ISOCS	In Situ Object Counting System
ISS	interim safe storage
JHA	job hazards analysis
LARADS	Laser-Assisted Ranging and Data System
LSFF	105-DR Large Sodium Fire Facility
MITUS	Mobile Integrated Temporary Utility System
MOC	Management of Change
MTCA	*Model Toxics Control Act*
NEPA	*National Environmental Policy Act of 1969*
PCB	polychlorinated biphenyl
PHMC	Project Hanford Management Contractor
PMII	Project Manager's Implementing Instructions
PPE	personal protective equipment
QAPjP	quality assurance project plan
RCRA	*Resource Conservation and Recovery Act of 1976*
RESRAD	RESidual RADioactivity dose model
RL	U.S. Department of Energy, Richland Operations Office

RWP	radiological work permit
S&M	surveillance and maintenance
SAP	sampling and analysis plan
SS HASP	site-specific health and safety plan
SSE	safe storage enclosures
Tri-Party Agreement	*Hanford Federal Facility Agreement and Consent Order*
TSCA	*Toxic Substances Control Act of 1976*
TSD	treatment, storage, and disposal
VSR	vertical safety rod
WAC	*Washington Administrative Code*

METRIC CONVERSION CHART

Into Metric Units

If You Know	*Multiply By*	*To Get*
Length		
inches	25.4	millimeters
inches	2.54	centimeters
feet	0.305	meters
yards	0.914	meters
miles	1.609	kilometers
Area		
sq. inches	6.452	sq. centimeters
sq. feet	0.093	sq. meters
sq. yards	.0836	sq. meters
sq. miles	2.6	sq. kilometers
acres	0.405	hectares
Mass (weight)		
ounces	28.35	grams
pounds	0.454	kilograms
ton	0.907	metric ton
Volume		
teaspoons	5	milliliters
tablespoons	15	milliliters
fluid ounces	30	milliliters
cups	0.24	liters
pints	0.47	liters
quarts	0.95	liters
gallons	3.8	liters
cubic feet	0.028	cubic meters
cubic yards	0.765	cubic meters
Temperature		
Fahrenheit	subtract 32, then multiply by 5/9	Celsius

Out of Metric Units

If You Know	*Multiply By*	*To Get*
Length		
millimeters	0.039	inches
centimeters	0.394	inches
meters	3.281	feet
meters	1.094	yards
kilometers	0.621	miles
Area		
sq. centimeters	0.155	sq. inches
sq. meters	10.76	sq. feet
sq. meters	1.196	sq. yards
sq. kilometers	0.4	sq. miles
hectares	2.47	acres
Mass (weight)		
grams	0.035	ounces
kilograms	2.205	pounds
metric ton	1.102	ton
Volume		
milliliters	0.033	fluid ounces
liters	2.1	pints
liters	1.057	quarts
liters	0.264	gallons
cubic meters	35.315	cubic feet
cubic meters	1.308	cubic yards
Temperature		
Celsius	multiply by 9/5, then add 32	Fahrenheit

1.0 INTRODUCTION

This document contains the removal action work plan for the 105-DR and 105-F Reactor buildings and ancillary facilities[1]. These buildings and facilities are located in the 100-D/DR and 100-F Areas of the Hanford Site, which is owned and operated by the U.S. Department of Energy (DOE), in Benton County, Washington. The 100 Areas (including the 100-D/DR and 100-F Areas) of the Hanford Site were placed on the U.S. Environmental Protection Agency's (EPA's) National Priorities List under the *Comprehensive Environmental Response, Compensation, and Liability Act of 1980* (CERCLA). The DOE has determined that hazardous substances in the 105-DR and 105-F Reactor buildings and four ancillary facilities present a potential threat to human health or the environment. The DOE has also determined that a non-time-critical removal action is warranted at these facilities.

Alternatives for conducting a non-time-critical removal action were evaluated in the *Engineering Evaluation/Cost Analysis for the 105-DR and 105-F Reactor Facilities and Ancillary Facilities* (DOE-RL 1998a). The engineering evaluation/cost analysis (EE/CA) resulted in the recommendation to decontaminate and demolish the contaminated reactor buildings (except for the reactor blocks) and the ancillary facilities and to construct a safe storage enclosure (SSE) over the reactor blocks. The recommendation was approved in an action memorandum (Ecology et al. 1998a) signed by the Washington State Department of Ecology (Ecology), EPA, and DOE. The DOE is the agency responsible for implementing the removal actions in the 105-D/DR and 105-F Areas. Ecology is the lead regulatory agency for facilities in the 100-D/DR Area, and EPA is the lead regulatory agency for facilities in the 100-F Area. The term "lead regulatory agency" hereinafter refers to these authorities. This removal action work plan supports implementation of the non-time-critical removal action.

1.1 PURPOSE AND OBJECTIVE OF THE REMOVAL ACTION WORK PLAN

The purpose of this removal action work plan is to establish the methods and activities to perform the following removal action functions:

- Complete decontamination and decommissioning (D&D) of the four ancillary facilities
  - 116-D exhaust air stack
  - 116-DR exhaust air stack
  - 117-DR Exhaust Filter Building
  - 119-DR Exhaust Air Sample Building

[1] The term "facilities" is used generically to encompass all the structures, buildings, tunnels, piping, etc., associated with the buildings. However, with regard to the reactor buildings (i.e., 105-F and 105-DR), the reactor blocks are not included in the removal action.

- Complete closure of the 105-DR Large Sodium Fire Facility (LSFF) treatment, storage, and disposal (TSD) unit
- Modify structures as necessary and construct interim safe storage (ISS) enclosures for the 105-DR and 105-F Reactor buildings
- Remediate waste sites within the reactor footprint or provide for deferral to the Remedial Action/Waste Disposal (RA/WD) Program (with approval from the lead regulatory agency)
- Manage and dispose of all waste generated during these actions.

This removal action work plan satisfies the requirement in the action memorandum (Ecology et al. 1998a) that DOE submit a removal action work plan outlining how compliance with applicable regulations (refer to Section 4.1) and enforceable schedule (refer to Section 5.1) will be achieved for cleanup of the 105-DR LSFF TSD unit, ancillary buildings demolition, and ISS of the reactor buildings. Additionally, it serves as the decommissioning plan and project management plan for the 105-DR and 105-F ISS Projects. This removal action work plan was prepared in accordance with Section 7.2.4 of the *Hanford Federal Facility Agreement and Consent Order* (Tri-Party Agreement) (Ecology et al. 1998b).

In addition to this removal action work plan, the action memorandum (Ecology et al. 1998a) specifies other deliverables that must be submitted by DOE to the lead regulatory agencies for review and approval. This removal action work plan describes the deliverables and provides a schedule for meeting the deliverables. The schedule includes interim actions that lead to completion of the associated Tri-Party Agreement milestones M-93-11 (Complete Interim Safe Storage of the 105-F Reactor Building by September 2004~~3~~) and M-93-16~~-T01~~ (Complete Interim Safe Storage of the 105-DR Reactor Building by September 2003~~5~~) (refer to Appendix A). The specific deliverables specified in the action memorandum and discussed in this removal action work plan include the following:

- Sampling and analysis plans (SAPs) for waste and soil characterization and disposal (Sections 4.2.1 and 4.3)
- Treatment plans if treatment is necessary prior to waste disposal in the Environmental Restoration Disposal Facility (ERDF) (Section 4.2.3)
- Verification SAPs for soil and below-grade structures (Sections 4.3 and 5.7)
- Cleanup verification package (Section 5.7).

The intent of this removal action work plan is to identify the basis and provide guidance for preparation of work packages for the project tasks. Using the most recent information concerning facility conditions, field-level work packages will be developed to direct work activities and instruct workers in the most applicable work methods. Existing procedures (as well as specifically developed instructions) will be used to perform and control the facility removal and disposal actions.

The 105-DR and 105-F ISS and ancillary building project schedule, which encompasses the work scope through project completion, presents the logical progression of events and estimated durations for each activity. The project schedule, included as Appendix A, is presented by fiscal year~~, and the resources with associated costs will be presented by activity~~.

1.2 SCOPE AND OBJECTIVES OF THE REMOVAL ACTION

The primary goal of CERCLA removal actions is to minimize or eliminate threats to public health or the environment caused by the presence of hazardous substances. The EE/CA for the 105-DR and 105-F Reactor buildings and ancillary facilities (DOE-RL 1998a) presents three alternative approaches for future facility management and the resulting levels of protection of public health and the environment that may be anticipated. Based on the evaluation, the ISS of the reactors and D&D of ancillary facilities were recommended as the most responsive approach to ensure protection of human health and the environment. The selection and approval of this approach are documented in the action memorandum (Ecology et al. 1998a) for the 105-F and 105-DR Reactor buildings and ancillary facilities.

The scope of the approved removal action includes the 105-F and 105-DR Reactor buildings (except for the reactor blocks) and the four ancillary facilities, all of which are described in Section 1.2. The 116-DR exhaust air stack, 117-DR Exhaust Filter Building, 119-DR Exhaust Air Sample Building and associated ducting/tunnels, and 116-D exhaust air stack are included in this removal action. Of the four ancillary buildings and facilities, two (116-DR exhaust air stack and 117-DR Exhaust Filter Building) are included within the boundaries of the 105-DR LSFF TSD unit, which must be closed under authority of the *Resource Conservation and Recovery Act of 1976* (RCRA). Although each of the four ancillary facilities is addressed in this document, each facility has a separate budget within the 105-DR ISS Project. The D&D of the 105-DR ancillary buildings/structures and TSD unit closure will occur before completion of the 105-DR ISS Project.

Stabilization, partial demolition, and disposal will reduce the potential hazards to public health or the environment that are currently posed by the 105-DR and 105-F Reactor buildings and ancillary facilities. Waste products generated by the D&D and safe storage activities will be separated into a variety of waste streams, each of which will be disposed at appropriate disposal facilities.

Below-grade structures will be removed to a minimum of 0.9 m (3 ft) below surrounding grade, and the remaining portion will either be removed or left in place, depending on whether cleanup standards can be achieved (see Section 4.1.1). Portions of the below-grade structures and soils that are above cleanup levels will be removed. In the event that large volumes of contaminated soil are encountered or removal of contaminated soil inhibits reactor safe storage activities, with concurrence of the lead regulatory agency, the removal of contaminated soils may be deferred to the RA/WD Program. The respective footprints of the reactor buildings, ancillary facilities, and the affected surrounding terrain will be backfilled after completion of the removal action if the cleanup standards (see Sections 4.1 and 5.7) are met. Site restoration will be coordinated with remedial actions and 100-F, 100-D, and 100-DR Area restoration actions. Characterization information for these areas will be generated to document the status of conditions at the conclusion of this project.

Based upon the selected alternative, the removal action objectives are as follows:

- To the extent practicable, reduce potential future releases of hazardous substances contained within facilities to acceptable protection levels established in applicable or relevant and appropriate requirements (ARARs)
- Protect workers from the hazards posed by these facilities
- Prevent adverse impact to cultural resources and threatened or endangered species
- Safely manage (i.e., treat or dispose) waste streams generated by the removal action
- Reduce or eliminate the need for future surveillance and maintenance (S&M) activities
- Coordinate clean closure of the TSD unit (at the 105-DR Reactor building) and place the 105-DR Reactor building into ISS
- Place the 105-F Reactor building into ISS
- Coordinate with the Bechtel Hanford, Inc. (BHI) RA/WD Program to address waste sites or activities that may interfere with the disposition of the 105-DR and 105-F Reactor buildings or ancillary facilities.

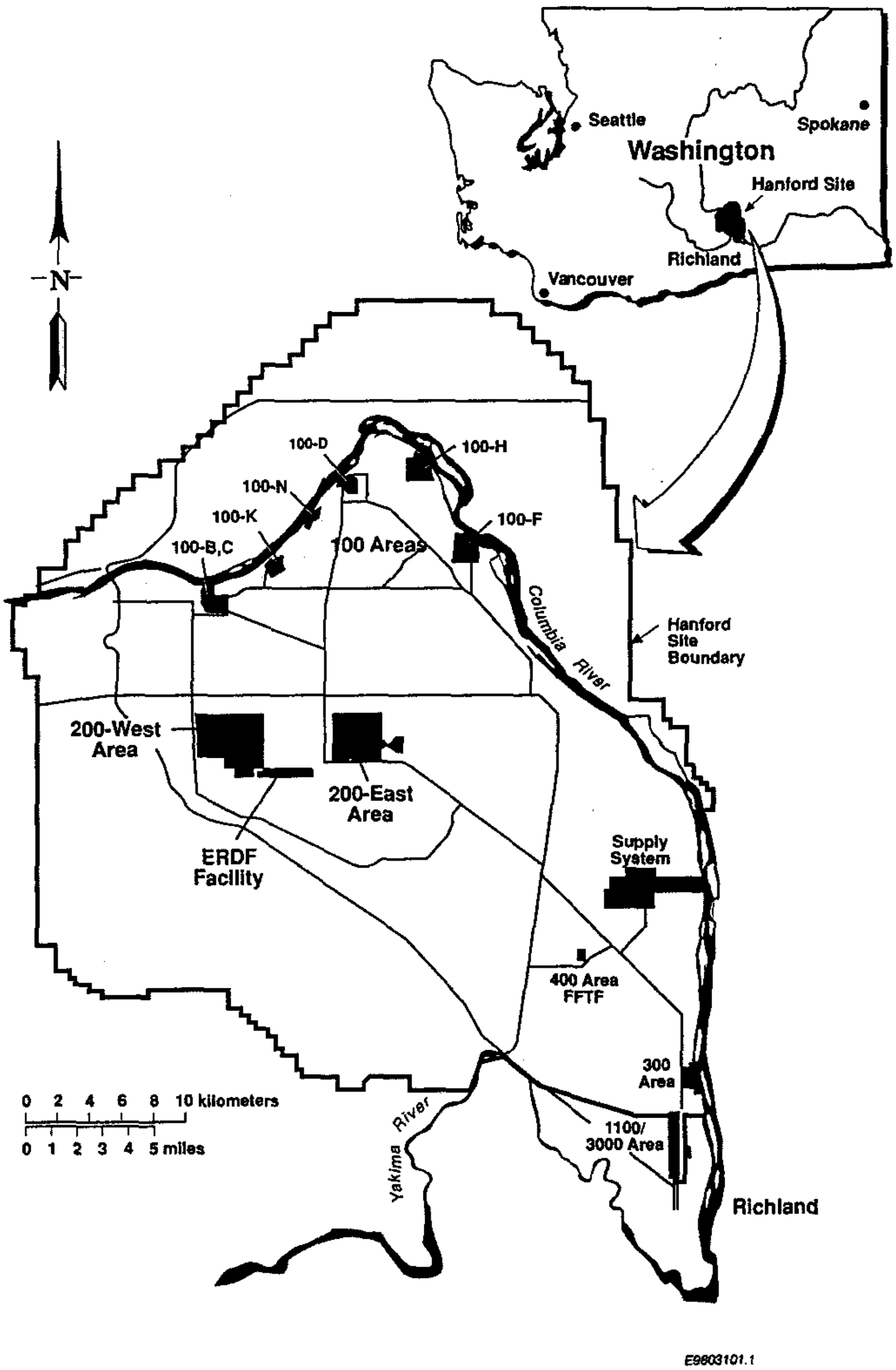
Section 1.3 provides a facility history and a description of each of the buildings and facilities covered by the removal action.

1.3 FACILITY HISTORY AND DESCRIPTION

1.3.1 Facility History

The reactor buildings, designated as the 105-F or F Reactor and the 105-DR or DR Reactor, were two of the nine water-cooled, graphite-moderated reactors constructed in the Hanford Site's 100 Areas (Figure 1-1) by the U.S. Government to support the plutonium production effort initiated in 1942. The reactors were located along the southern bank of the Columbia River in southeastern Washington State. Construction of the 105-F Reactor began in December 1943; operations began in February 1945, and the reactor was shut down in June 1965. Construction of the 105-DR Reactor began in December 1947; operations began in October 1950, and the reactor was shut down in December 1964. The 105-F and 105-DR Reactors were placed in final shutdown mode and declared as surplus by DOE, and D&D will be performed on each of the reactors.

Figure 1-1. Hanford Site Map.



On May 16, 1985, DOE published in the *Federal Register* (50 FR 20489) a "Notice of Intent to Prepare an Environmental Impact Statement on Decommissioning the Eight Shutdown Production Reactors Located at the Hanford Site Near Richland, Washington."

In December 1992, a final environmental impact statement (FEIS) (DOE 1992) was issued that analyzed five decommissioning alternatives for the eight shutdown reactors. The FEIS recommended safe storage of the reactors, followed by deferred one-piece removal of the reactor blocks. In September 1993, a Record of Decision (58 FR 48509) documented the selection of the safe storage/deferred removal alternative for all the 100 Areas' surplus production reactors, with the exception of the 105-N Reactor. The 105-N Reactor was shut down with no intent for reuse in 1991 (DOE-RL 1995) and therefore was not included in the FEIS (DOE 1992).

1.3.2 Facility Descriptions

1.3.2.1 105-F and 105-DR Reactor Buildings. The 105-F and 105-DR Reactor buildings are reinforced-concrete and concrete block structures with steel framing. The lower levels of the buildings and central portions surrounding the reactor blocks are constructed of reinforced-concrete walls, 0.9 to 1.5 m (3 to 4.9 ft) thick. Each reactor building includes the following components:

- Reactor block
- Fuel storage basin (FSB)
- Inner and outer horizontal control rod (HCR) rooms
- Vertical safety rod (VSR) winch level
- Front-face work area
- Fans and ducts for air ventilation and recirculating inert gas systems, including water cooling systems
- Supporting offices, shops, and laboratories.

Figures 1-2 and 1-3 show the layout of the buildings at ground level, including some of the areas described above. The outside dimensions of the 105-F Reactor building are approximately 82.7 m by 95.8 m by 28.4 m (271.3 ft by 314.3 ft by 93.2 ft) high. The outside dimensions of the 105-DR Reactor building are approximately 82.7 m by 95.8 m by 32 m (271.3 ft by 314.3 ft by 105 ft) high. The existing roof panels were removed from the process area, D elevator, and front-face work area in 1994-1995 and were replaced with steel roof decking secured to existing roof framing and concrete walls. The new steel roof decking was covered with foam and two applications of silicone rubber. The 105-F FSB and adjoining transfer bay roofs were replaced with a similar roof in 1993.

Figure 1-2. Floor Plan Layout at Ground Level for the 105-DR Facility.

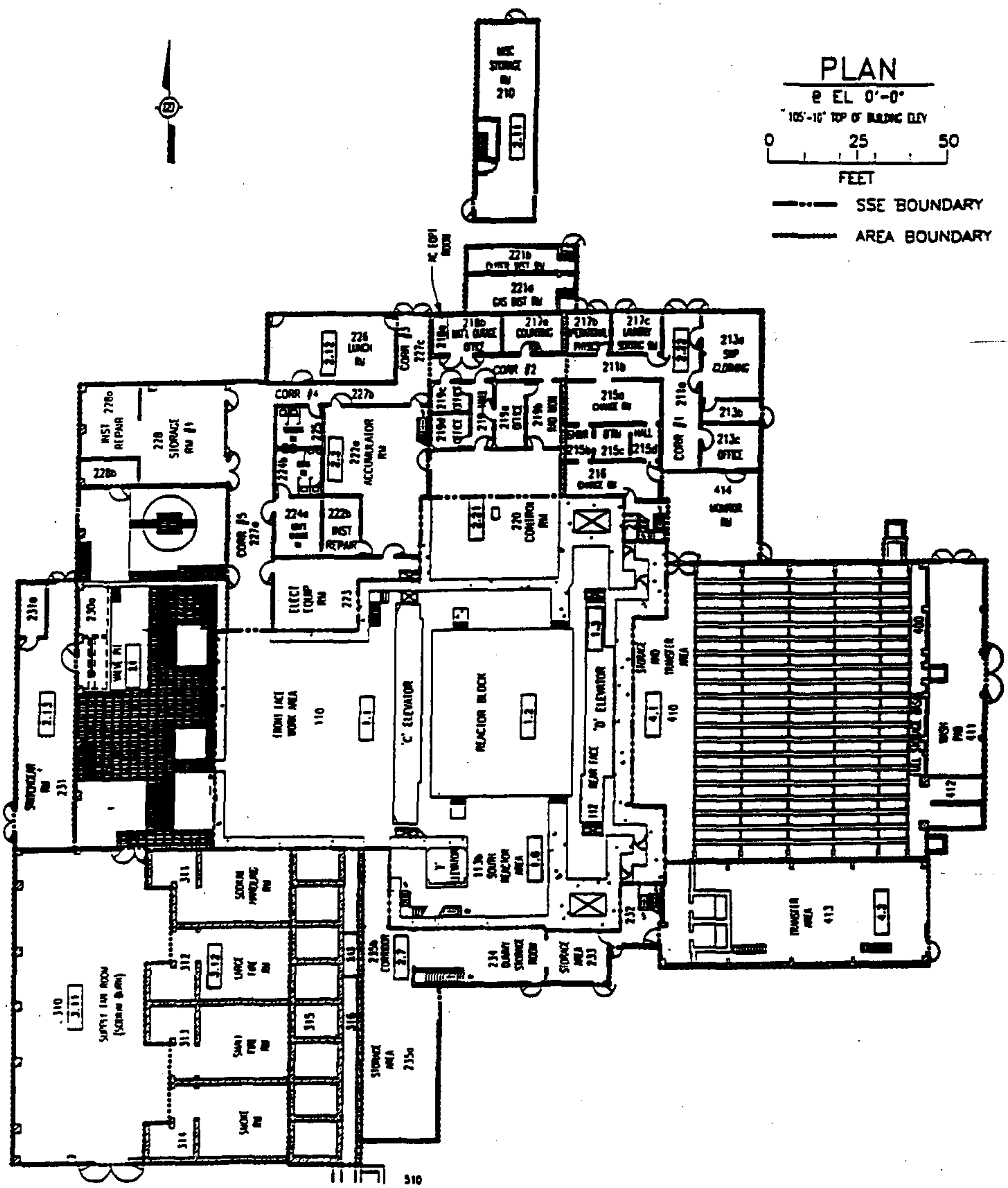
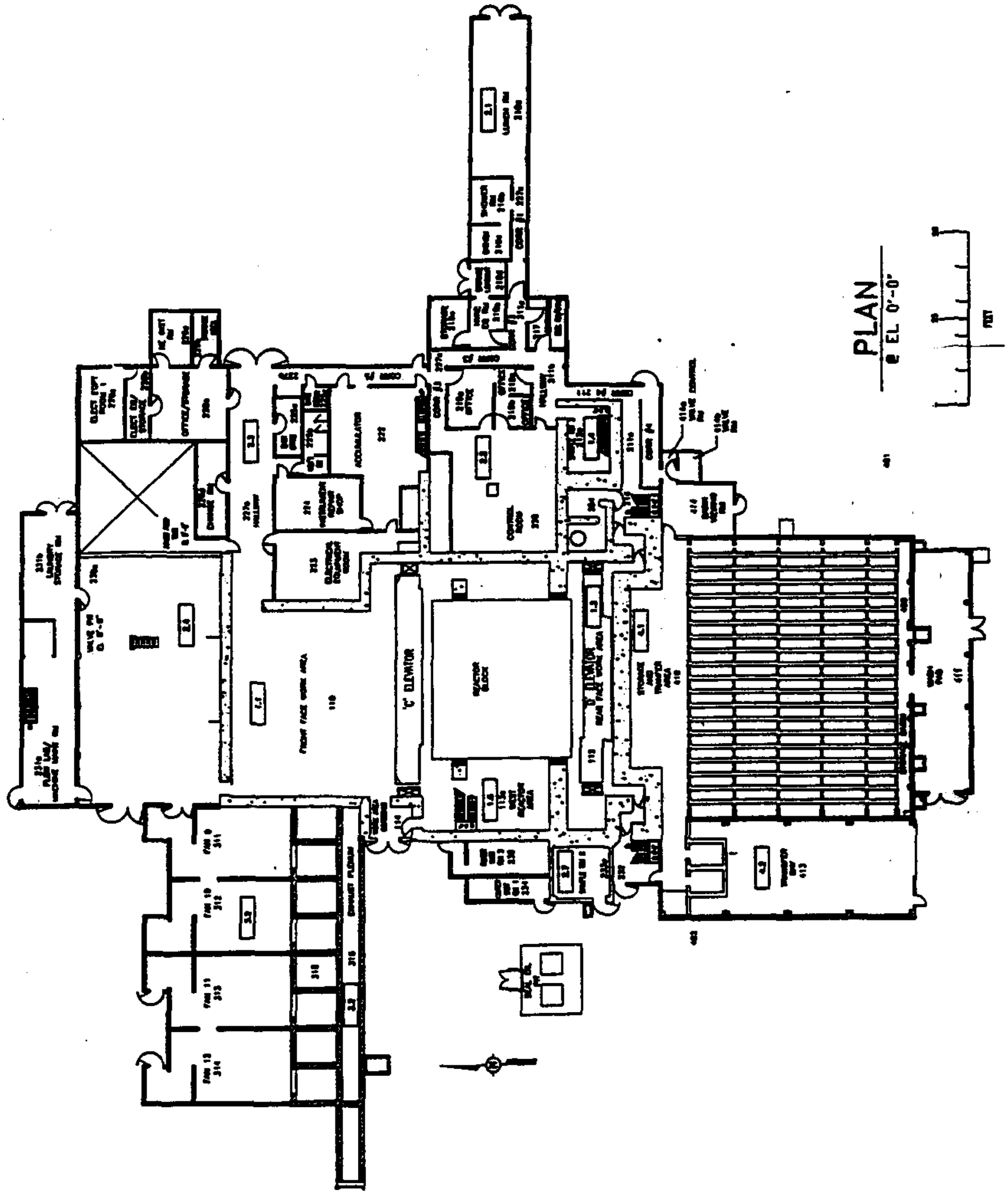


Figure 1-3. Floor Plan Layout at Ground Level for the 105-F Facility.



The internal layouts of retired 100 Areas' reactors are typically defined in terms of areas. The areas are identified as follows:

- Module 1 (the general ancillary area of the 105-F and 105-DR Reactor buildings that is located outside of the shield walls)
- Module 2 (the area within the shield walls, including the reactor blocks)
- FSB areas
- Ancillary buildings.

Figures 1-4 and 1-5 provide the layout for the 105-F and 105-DR Facilities, depicting the relationships of these four areas. A more detailed description follows.

1.3.2.1.1 Module 1. Module 1 provided ancillary support areas during former reactor *operations. These support areas included office areas, the reactor control room, tool storage* rooms, restrooms, cooling water influent areas, change rooms, ventilation equipment areas, electrical systems areas, and other infrastructure support.

1.3.2.1.2 Module 2. Module 2 is the area located inside the shield walls, including the reactor blocks. Figures 1-6 and 1-7 provide a plan view of Module 2 and what will subsequently be the SSEs. In addition to the reactor blocks, areas and rooms within Module 2 include the inner rod rooms, front face work areas, 3X ball systems, VSR winch assembly mechanism areas, laboratories, and other support areas.

Reactor Blocks. The reactor blocks are located near the center of the buildings within Module 2. Each reactor block consists of a 10.2-m by 12.5-m by 12.5-m (38.5-ft by 41-ft by 41-ft) graphite moderator stack, encased in a 20.3-to 25.4-cm (8- to 10-in.)-thick overlapping cast-iron thermal shielding; 132-cm (52-in.)-thick welded biological shields consisting of alternating layers of masonite and steel on the four sides (excluding the bottom of the stacks); and unwelded, stair-step labyrinth seal shields on top. Figure 1-8 provides a three-dimensional illustration of the reactor blocks. The blocks rest on concrete foundations. The main components of each block are as follows:

- Reactor moderator stack (an assembly of graphite blocks cored to provide channels for the process tubes, control rods, and other equipment)
- Aluminum process tubes that held the uranium metal fuel elements and provided channels for cooling water
- Control rods, gun barrels, monitoring equipment, experimental test holes, etc.

Figure 1-4. Layout of the 105-DR Facility (Module 1, Module 2, Fuel Storage Basin, and Ancillary Buildings).

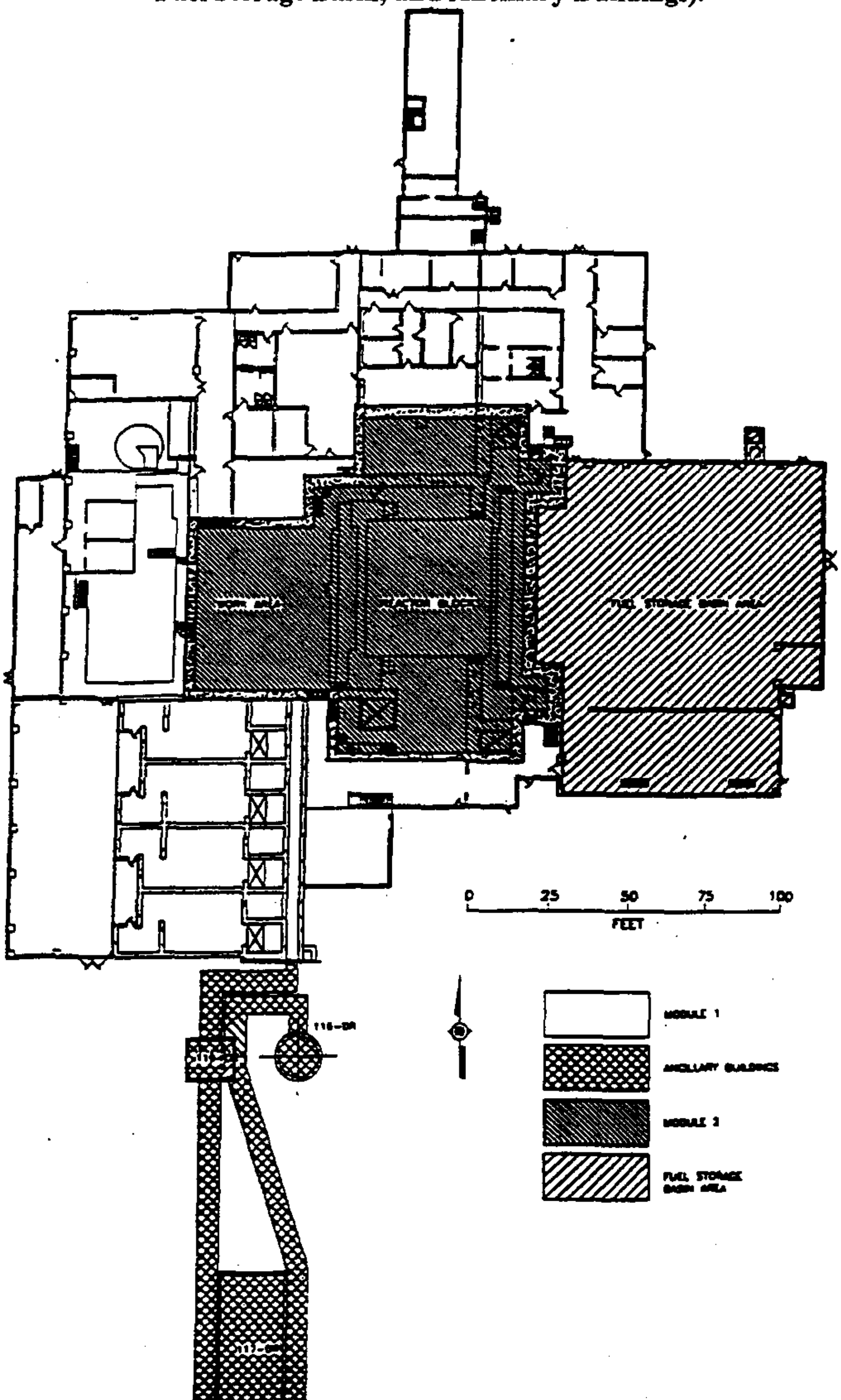


Figure 1-5. Layout of the 105-F Reactor Building (Module 1, Module 2, and Fuel Storage Basin).

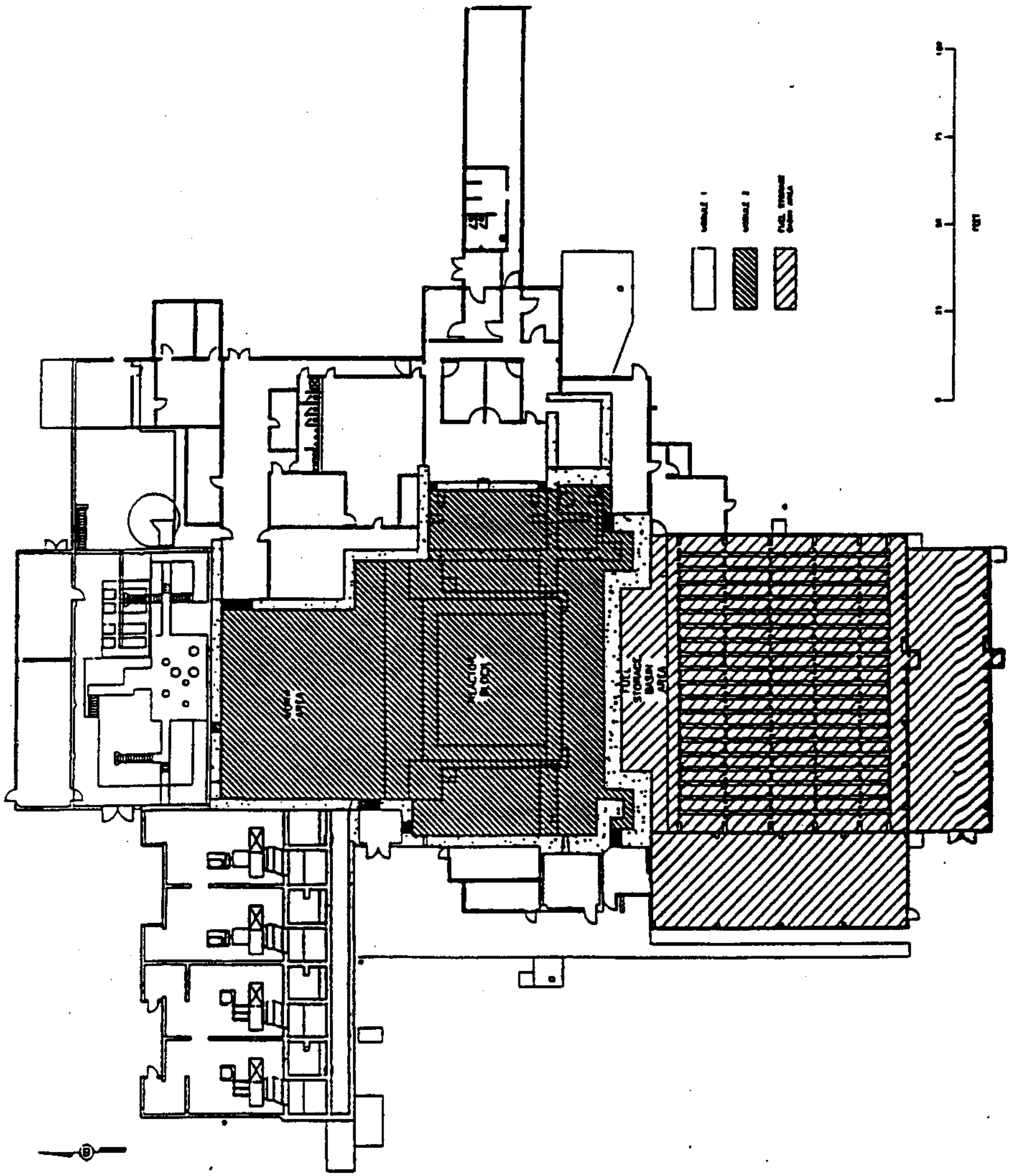


Figure 1-6. General Plan of the 105-DR Safe Storage Enclosure at Ground Level.

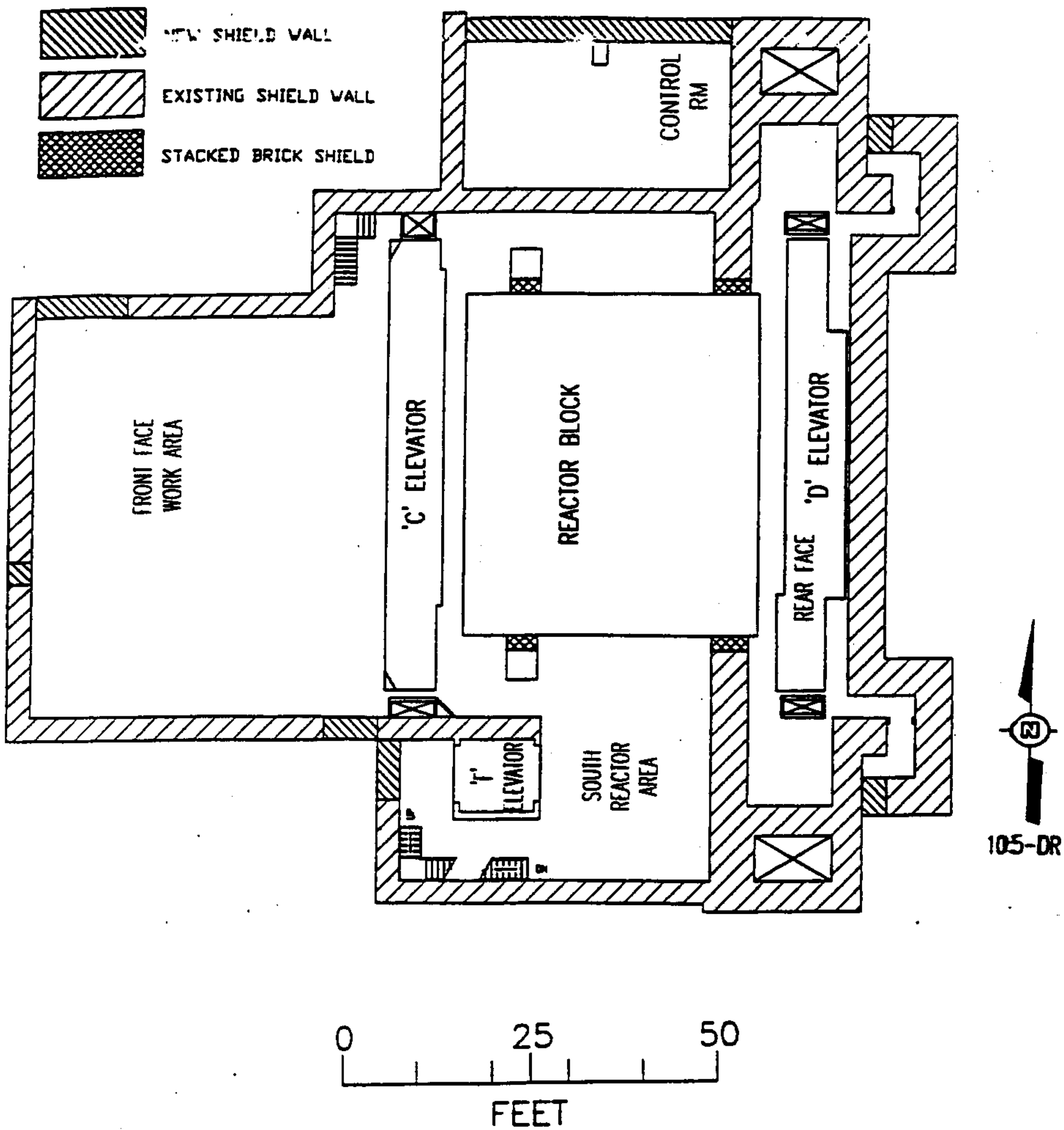


Figure 1-7. General Plan of 105-F Safe Storage Enclosure at Ground Level.

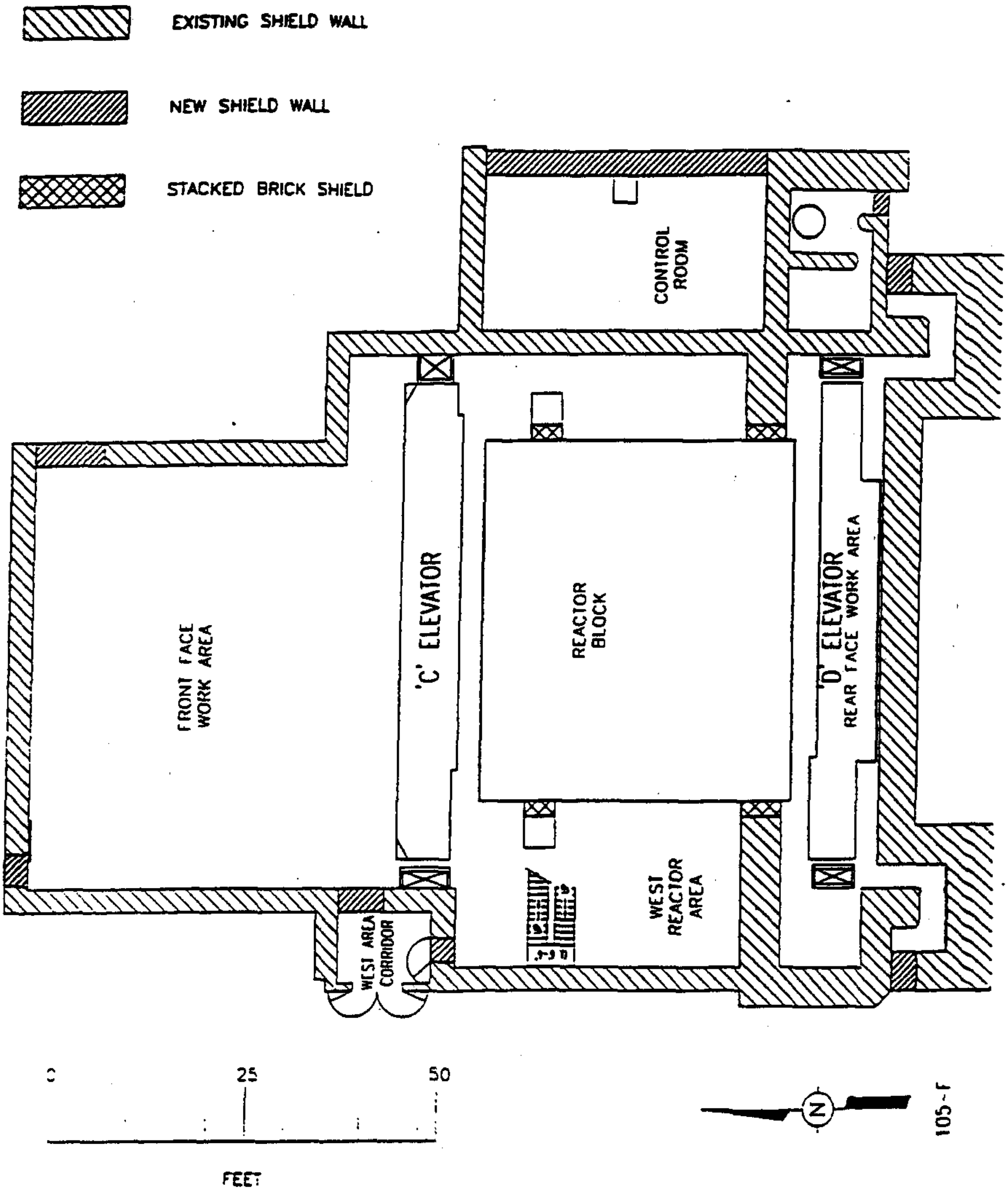
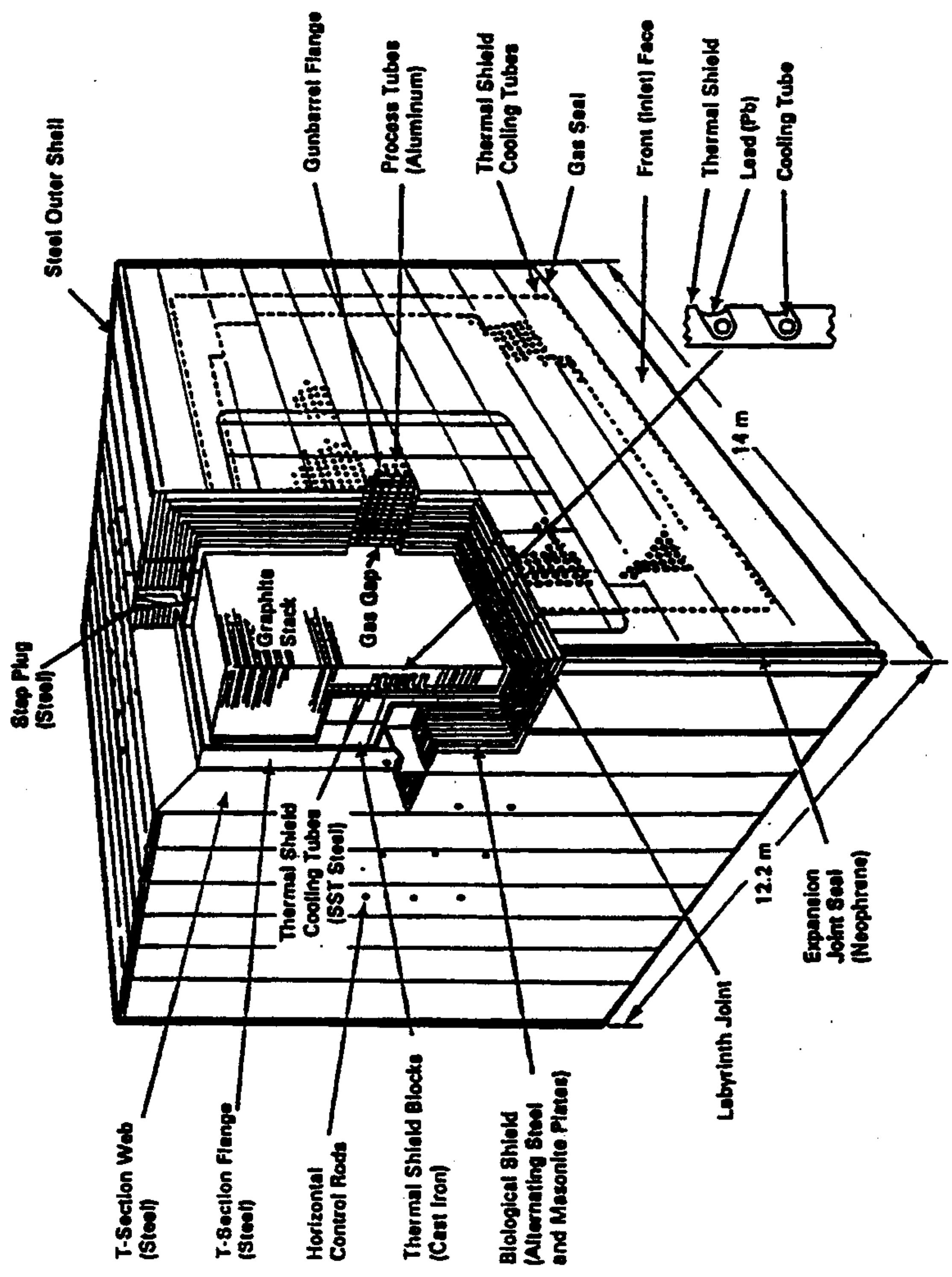


Figure 1-8. Reactor Block Construction.



- Thermal and biological shields

- Welded steel-plate box that enclosed the biological shield and served to confine the inert gas atmosphere within the reactor.

The HCRs entered the reactors from the left side (when facing the reactor front face), and the VSRs and 3X ball system penetrations entered the reactors from the top. The 3X ball system was an emergency shutdown system consisting of neutron-absorbing balls that could be released into the core from hoppers above the reactors. The experimental test facility used penetrations in the right side of the reactor blocks for irradiation and detection studies. The HCRs had inner and outer rod rooms. The inner rod rooms are located within the massive shield walls of the buildings; the walls provided protection to workers when the rods were removed from the reactors during operation. Fuel discharge and storage areas are located directly behind the rear of the reactors.

All penetrations through the outer shielding of the blocks (e.g., the process tubes, HCR/VSR/3X ball channels, and instrumentation/experimental channels) are provided with cover-gas-seal-designed tubes/sleeves/thimbles to maintain cover-gas containment. The channel tubes, if left sealed and intact, eliminate outside pathways to the inner graphite stack areas.

Reactor Block Shutdown. During final shutdown of the reactors, a number of deactivation procedures were performed to contain contamination within the reactor blocks. After confirmation that all of the tubes had been discharged, process caps were installed at the front and rear of the reactor blocks (GE 1965). The crossheader valves were tagged closed. After all of the cover-gas lines were drained, the drain valves were closed. All gas sample lines were closed at the first valve of the primary system and at the inlet and outlet sample chambers. All water lines were drained and all HCRs and VSRs were left in the "full-in" position. The balls were vacuumed from the hoppers, and the majority were stored in metal containers with desiccant added. Position and temperature indicators were isolated and pressure-sensing lines were drained and blown dry. The line valves were closed, capped, or plugged.

Special samples were discharged from the numerous test holes in the reactor blocks. Test holes with water were drained and purged with air until the water was removed. Shielding was reinstalled in any experimental channels (the 105-DR Reactor block has seven experimental channels).

Several areas of the reactor block require additional protective measures to ensure 75 years of safe storage. These areas were identified as the VSR gaskets and the HCR openings. These areas will be stabilized using appropriate and approved design methods similar to those used on the 105-C Reactor ISS Project.

1.3.2.1.3 Fuel Storage Basin Areas.

105-F Fuel Storage Basin. The 105-F FSB area, located on the south side of the 105-F Reactor building, served as an underwater collection, storage, and transfer facility for the irradiated fuel elements discharged from the reactor. The FSB area consists of the fuel element discharge pickup area, which is located adjacent to the reactor rear face; the fuel storage area, which is the basin proper; the fuel transfer area, which includes the fuel transfer pits; and the wash pad area used to decontaminate fuel-handling equipment. The storage basin is approximately 22 m by 25 m by 6.1 m (72.2 ft by 82 ft by 20 ft) deep. The FSB area is approximately 822.7 m^2 (8,852 ft^2). The transfer bay is located west of the FSB and served as a railcar cask-loading area for transferring fuel from the FSB. The transfer bay also has a 3X ball washer located in the southwest corner. Figure 1-9 shows the FSB layout.

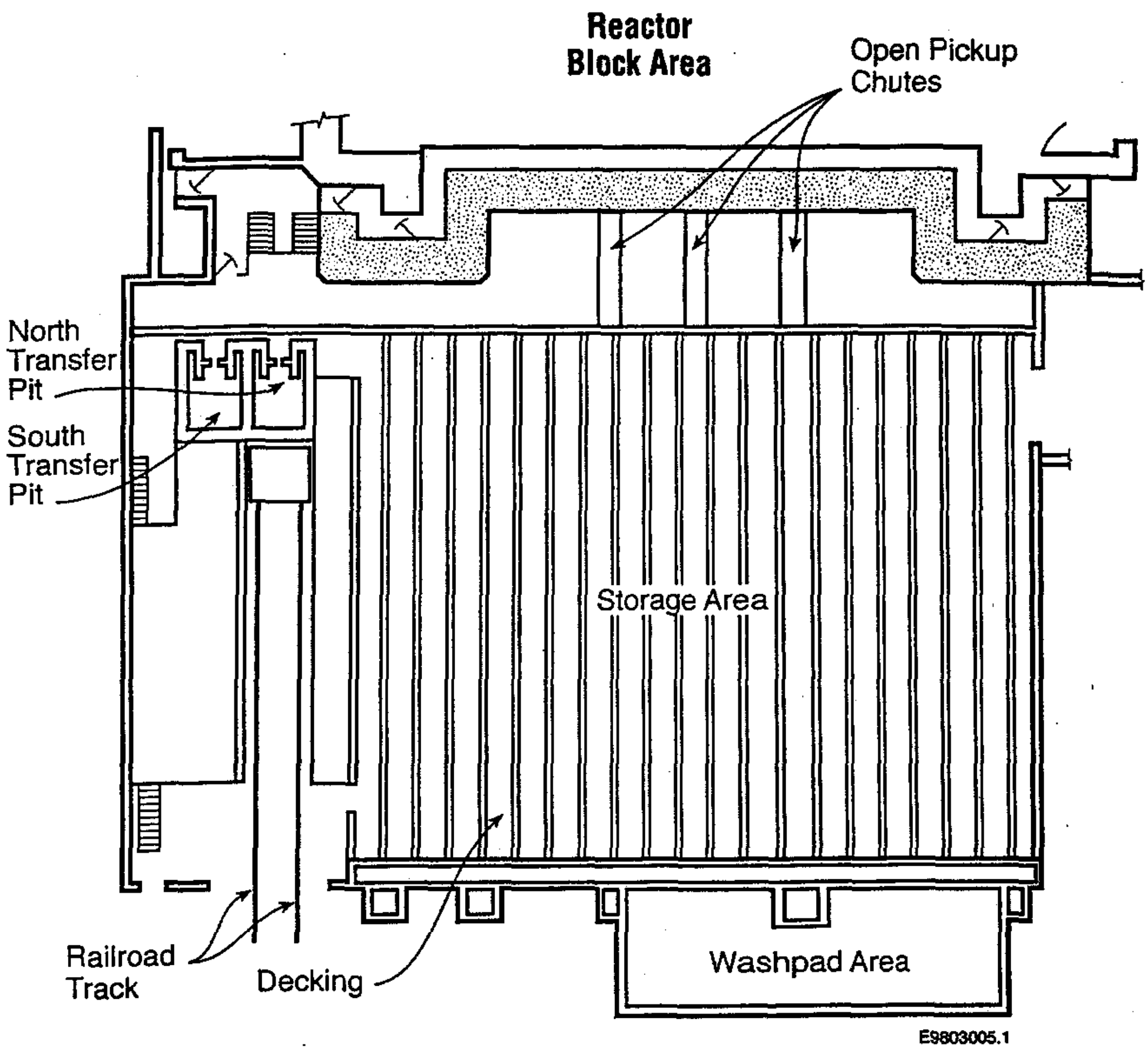
The deactivation of the FSB at the 105-F Reactor building occurred in 1970. Liquid was pumped from the basin until 0.6 m (2 ft) of water remained. Within this bottom 0.6 m (2 ft) of water, sediment/sludge and miscellaneous items were left behind. Items known to have been left behind in the basin include fuel buckets, fuel spacers, process tubes, tongs, wooden floor decking, and monorail pieces. After the FSB was partially drained, fine stream-bed sand backfill was placed into the remaining 55 m (18 ft) of the basin. The specific characteristics of the backfill placed in the 105-F FSB are unknown; therefore, backfill characterization sampling and analysis for hazardous chemicals and radiological material using EPA protocol was performed from August through September 1991. The backfill characterization report (UNI 1981), issued in October 1993, concluded the following:

- The top 3.1 m (10.2 ft) of backfill has minimal contamination.
- The next 2.1 m (6.9 ft) is suspected to be radiologically contaminated in some locations.
- The bottom 0.9 m (3 ft) is contaminated with material, equipment, sediment layer, and possibly fuel elements, mainly residing close to the basin floor.

The 105-F FSB inventory is described in greater detail in the *Final Hazard Classification and Auditable Safety Analysis for the 105-F Reactor Interim Safe Storage Project* (BHI 1998c).

105-DR Fuel Storage Basin. The 105-DR FSB area, located on the east side of the 105-DR Reactor building, served as an underwater collection, storage, and transfer facility for the irradiated fuel elements discharged from the reactor. The FSB area consists of the fuel element discharge pickup area, located adjacent to the reactor rear face; the fuel storage area; the basin proper; the fuel transfer area including the fuel transfer pits; and the wash pad area used to decontaminate fuel-handling equipment. The storage area dimensions are approximately 22 m by 24.7 m by 6.1 m (72.2 ft by 81 ft by 20 ft) deep. The FSB area is approximately 822.7 m^2 (8,852ft^2). The FSB and the fuel transfer pits have been drained and cleared of debris and sediment, and a fixative has been applied to the lower portions of the contaminated surfaces (UNI 1986). An asphalt emulsion fixative was applied on the bottom 2.44 m (8 ft) of the FSB walls and the FSB floor. The FSB is covered with fire-resistant-coated wood planking (UNI 1986).

Figure 1-9. Typical Fuel Storage Basin Area Layout.



1.3.2.1.4 Buildings Ancillary to the 105-F and 105-DR Reactor Buildings. The 116-F exhaust stack and stack foundation was decontaminated and demolished using explosives in September 1983 (UNI 1985). A trench was excavated west of the stack between the 115-F Gas Recirculation Building and 117-F Filter Building burial sites (UNI 1985). Demolition charges caused the stack to fall into the excavated trench. The rubble from the stack foundation was pushed into the trench with heavy equipment, and the trench was covered with clean fill to a depth of at least 1 m (3.3 ft) (UNI 1985). Allowable residual contamination level calculations on the stack and foundation indicated that doses from the rubble were below radiological control release limits, so the rubble was left in place (UNI 1985). *ARCL Calculations for Decommissioning the 116-F Stack* (UNI 1985) describes the levels of contamination and estimated dose scenarios in greater detail. The previously existing 100-F ancillary buildings are shown in Figure 1-10.

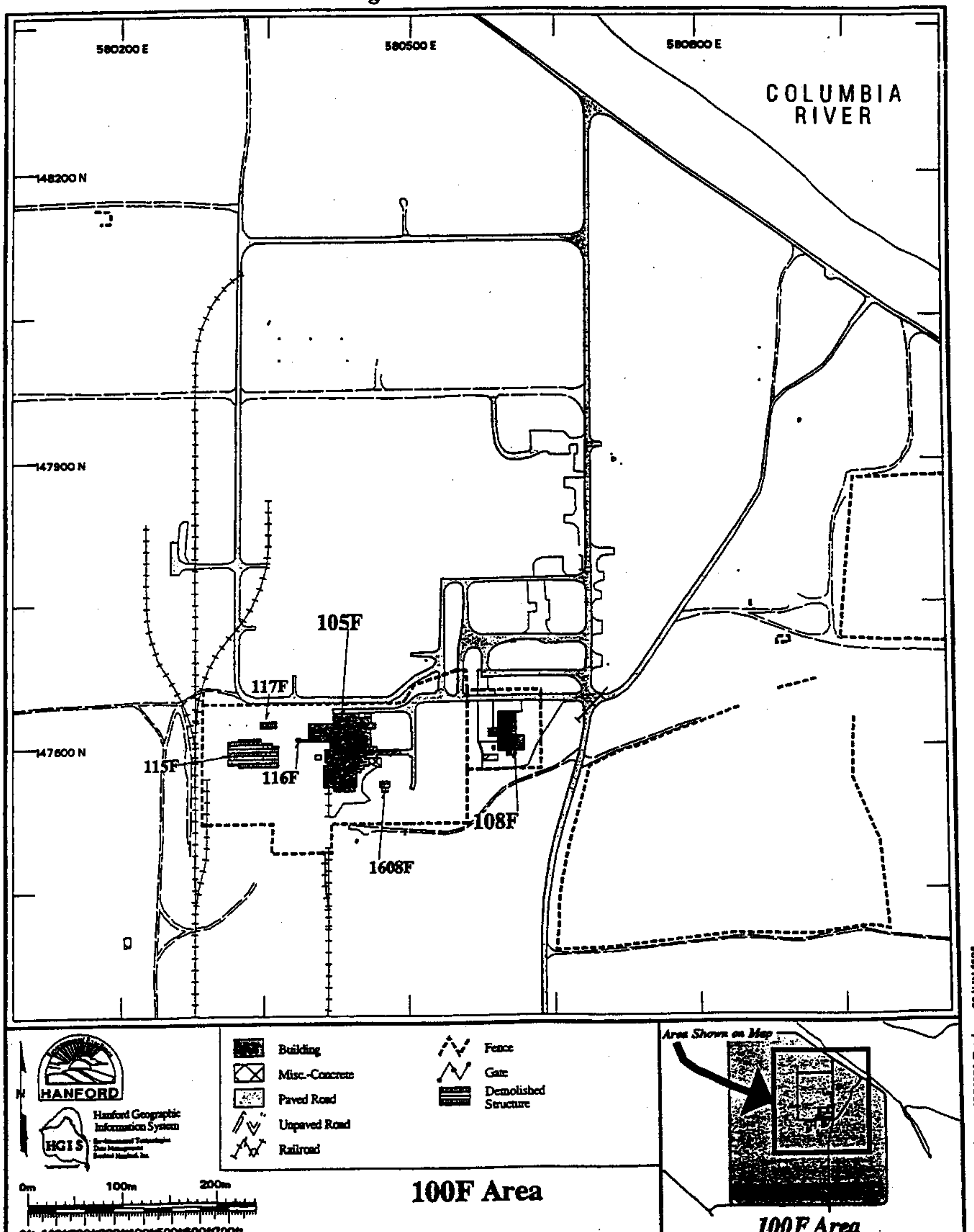
The D&D for the 1608-F and 1608-DR lift stations was performed in fiscal years 1986-1987 (RHO 1987a, 1987b). The lift stations were built as separate buildings, 1608-F and 1608-DR, outside of the footprint of the 105-F and 105-DR Reactor buildings. The 1608-F and 1608-DR lift stations were effluent water pumping stations that served as sumps to collect radioactively contaminated liquid wastes from the buildings. The liquid wastes were collected in the sumps for final disposal by pumping into the effluent system. All pipes entering the buildings were isolated and plugged using the dry-pack grout method to prevent water from draining into the ancillary buildings after demolition of the 1608-F and 1608-DR Buildings (RHO 1987a, 1987b).

1.3.2.2 Ancillary Buildings Covered by the Removal Action. In addition to the 105-F and 105-DR Reactor buildings, four ancillary buildings and associated facilities (e.g., underground tunnels and ducting) are covered by the removal action: 116-DR exhaust air stack, the 117-DR Exhaust Filter Building, the 119-DR Exhaust Air Sample Building and associated ducting/tunnels, and the 116-D exhaust air stack. The 116-DR exhaust air stack and the 117-DR Exhaust Filter Building are encompassed within the boundaries of the RCRA TSD unit. A brief history of the TSD unit and specific facility descriptions follow.

Between 1972 and 1986, the southwest portion of the 105-DR Reactor building was used as a research laboratory known as the 105-DR LSFF. The LSFF occupied the former ventilation supply fan room and was established to provide a means of investigating fire and safety aspects associated with large sodium or other metal alkali fires.

The LSFF had also been used for the storage and treatment of alkali metal dangerous waste. Additionally, the Fusion Safety Support Studies Programs sponsored intermediate-size safety reaction tests in the LSFF with lithium and lithium-lead compounds. The LSFF, a RCRA TSD unit, was partially clean closed in 1995, as documented in the *Closure Report for the N Reactor Facility* (DOE-RL 1995). "Clean closure" means that dangerous wastes (or dangerous waste constituents or residues) have been removed or decontaminated to ensure that levels do not exceed the numeric cleanup levels calculated using residential-exposure assumptions according to the "Model Toxics Control Act-Cleanup" regulations (*Washington Administrative Code* [WAC] 173-340) or closure performance standards (WAC 173-303-610[2][a][ii]).

Figure 1-10. 100-F Area.



The closure also must be in a manner that minimizes or eliminates post-closure escape of dangerous waste constituents. The portions of the 105-DR LSFF that have been clean closed include the 105-DR supply fan room (demolished), 1720-DR LSFF sodium storage building (demolished), the 116-DR exhaust air stack (demolished), and a majority of the surrounding soils. The 117-DR Exhaust Filter Building and associated ducting/tunnels and soils have not yet been closed. The locations of these facilities are shown in Figure 1-11.

1.3.2.2.1 116-DR Exhaust Air Stack. The 116-DR exhaust air stack, located on the south side of the 105-DR Reactor building, is a reinforced-concrete, monolithic, 61-m (200-ft)-high structure that discharged ventilation exhaust air to the atmosphere. The stack rests on a double octagon-shaped base that extends 5.3 m (17.4 ft) below grade. The stack is 4.9 m (16.1 ft) in diameter at its base and was fed by below-grade air exhaust ducts from the 117-DR Exhaust Filter Building, which filtered exhaust air generated in the 105-DR Reactor building and the experimental sodium burn facility located in the fan room. The stack was demolished in August 1999.

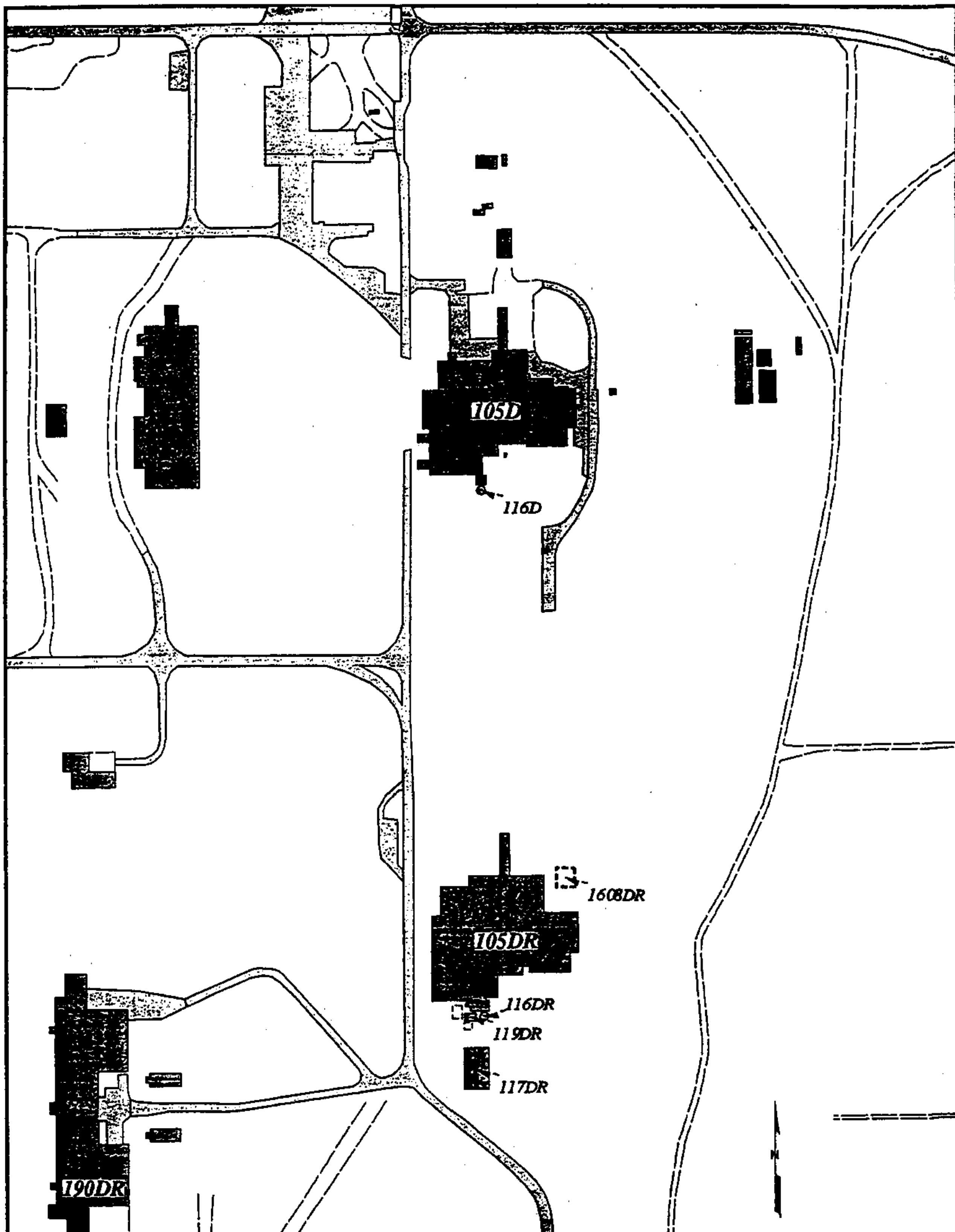
1.3.2.2.2 117-DR Exhaust Filter Building. The 117-DR Exhaust Filter Building is a reinforced-concrete structure, 18 m (59.1 ft) long, 11.9 m (39 ft) wide, 10.7 m (35.1 ft) high, and 2.4 m (7.9 ft) above grade. The building filtered ventilation air from the 105-DR Reactor building, including the experimental sodium burn facility, before discharging the air to the atmosphere through the 116-DR exhaust air stack.

1.3.2.2.3 119-DR Exhaust Air Sample Building. The 119-DR Exhaust Air Sample Building is a small, prefabricated metal building, 109.7 m^2 (1,180.4 ft^2), that housed most of the instrumentation for the exhaust air system. A sample stream of the exhaust air was routed through a continuous air monitoring system in the building to test for radioactivity. The building was demolished in July 1999.

1.3.2.2.4 Associated Exhaust Ducting/Tunnels. Ducting and tunnels connect the 105-DR exhaust fan room to the 119-DR Exhaust Air Sample Building, 117-DR Exhaust Filter Building, and 116-DR exhaust air stack. Most portions of the tunnels/ducting are located below grade, but some portions can be seen above grade.

1.3.2.2.5 116-D Exhaust Air Stack. The 116-D exhaust air stack, located on the south side of the 105-D Reactor, is a reinforced-concrete, monolithic, 61.0-m (200-ft)-high structure that discharged ventilation exhaust air to the atmosphere. The stack rests on a double octagon-shaped base that extends 5.3 m (17.4 ft) below grade. The stack is 4.9 m (16.1 ft) in diameter at its base and was fed by below-grade air exhaust ducts from the 117-D Exhaust Filter Building (D&D was performed in 1996), which filtered exhaust air generated in the 105-D Reactor. The stack was demolished in August 1999.

Figure 1-11. 100-D/DR Area.



2.0 REMOVAL ACTION

The objective of the 105-DR and 105-F ISS Projects is to place the 105-DR and 105-F Reactor buildings into ISS for a period of up to 75 years and to demolish the ancillary buildings. These actions will reduce the potential for release/exposure of hazardous and radioactive materials to workers, the public, and the environment. The ISS projects will also reduce periodic S&M costs incurred from maintenance of the degrading buildings.

2.1 REMOVAL ACTION WORK ACTIVITIES

The following sections provide a general description of how work activities will be performed for the 105-DR and 105-F ISS Projects. The general scope of work involved to implement this removal action includes the following activities:

- Removing hazardous substances (chemical and radiological)
- Removing facility equipment and miscellaneous piping
- Dismantling various facility structures
- Disposing of waste
- Constructing the SSE
- Performing verification sampling
- Preparing the project closeout report
- Preparing the cleanup verification package.

This work scope will be accomplished by completing the activities described in the following subsections.

2.1.1 Site Mobilization and Preparation Work

Upon initiation of the 105-DR and 105-F ISS Projects, personnel will be mobilized and consumables and required equipment will be procured. The first activities to be performed will include mobilizing manual personnel and trailers to support the project activities. Field Support personnel will also terminate and/or verify termination of all 105-F and 105-DR Reactor building services and utilities. Utilities currently servicing the facilities are electrical power and sanitary sewer. Electrical systems that will be used throughout the 105-DR and 105-F ISS Projects are discussed in further detail in Section 2.3.1.

Concurrent with these activities, waste segregation and staging areas will be set up (within the area of contamination (AOC) or at an onsite location) to facilitate transportation of the material for recycling or disposal in accordance with Section 4.2. Supervisor trailers, lunch trailers, change trailers, office trailers, mobile shower trailers, and restroom facilities will also be mobilized at the site to prepare for D&D activities. Electricity will be connected from an outside line or generator and temporary lighting will be installed. Occupational Safety and Health

Administration concerns, (e.g., fall protection, guarding, and electrical) will be managed as they are identified.

2.1.2 Characterization

Characterization will be conducted by the sampling and analysis of materials and radiological surveys throughout the 105-DR and 105-F ISS Projects and D&D of the ancillary buildings. Sampling and analysis will be used to identify radiological and hazardous conditions that will be encountered during facility operations, to specify health and safety requirements, and to determine the waste streams that will be generated. These technical services will also be used to characterize waste for treatment and disposal and to verify facility and area conditions at various phases of project completion. Section 4.3 describes characterization in further detail.

2.1.3 Decontamination and Demolition

Work activities, in general, will begin in the radiological controlled areas and will move progressively inward toward the contamination areas surrounding the reactor blocks. One of the first types of work activities inside the building will include conducting radiological surveys. Radiological surveys will be performed using hand-held and large-area detection equipment, which will be augmented with the Laser-Assisted Ranging and Data System (LARADS) for recordkeeping during surveys. Additionally, an advanced characterization system, consisting of an AIL GammaCam™ M31 gamma-ray/video imaging system for the location of hot spots, a Canberra In Situ Object Counting System (ISOCS) gamma-ray spectrometric system, and data-logging contamination monitoring equipment, may be deployed for the rapid, economical, and accurate characterization of materials for radiological contamination.

After an area has been surveyed and the radiological conditions have been established, biological cleanup and general housekeeping will commence, which involve removing loose biological feces and rubble and sweeping and vacuuming floors. Section 4.2 discusses how waste will be managed. At this time, asbestos-containing materials (ACMs) and presumed ACMs will be removed. The ACM typically consists of insulation for piping, floor tiles, cement asbestos board, etc. Insulation on piping will be removed as Class I asbestos work, and nearly all other ACM in the facilities will be removed as Class II (e.g., floor tiles and cement asbestos board). Asbestos work, air monitoring, and worker safety requirements will conform to BHI procedures for ACM removal and will be described in detail in the asbestos abatement work plan.

Lead bricks and sheeting, polychlorinated biphenyls (PCBs) in oil from motors and light ballasts, sodium dichromate from water system treatment, mercury in lighting components and switches, and other hazardous materials will be removed and disposed as hazardous or mixed waste or will be recycled consistent with guidelines found in Section 4.2.

Most of the loose, accessible radiological contamination will be either removed or fixed in place, depending upon the levels, accessibility, complex shapes (e.g., grating) and type of

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contamination found. Some equipment/piping will be removed, and loose contamination will be wiped or vacuumed. If loose contamination remains after the initial decontamination effort, unless the area will be inaccessible after completion of the ISS Project, or if the building configuration or conditions make removal of loose contamination impractical, then the contamination may be fixed in place, as required.

Activities that have a potential to emit radiological emissions are discussed in Appendix B. The above-grade structure outside the existing shield walls, including the siding and structural framing above the shield walls, will be demolished using standard demolition techniques. Steel will be segregated for salvage if economically feasible, unless determined to be contaminated. Included in the removal operation are the building exhaust fan room, supply fan room, maintenance shops, offices, control room, outer rod room, fuel storage areas, inlet water tunnels, change rooms, corridors, machinery rooms, valve pit, and other miscellaneous spaces currently located outside the shield wall, as shown in Figures 1-2 and 1-3. (Note: The supply fan room in the 105-F Reactor building was demolished as part of an earlier project.) Below-grade areas of the buildings that meet the cleanup criteria (defined in Sections 4.1, 4.4, and 5.7) will be left in place, and voids will be backfilled with clean debris or soil.

Decontamination necessary to allow removal of demolition equipment from contamination areas will be accomplished using standard industry practices and best management practices. Gross equipment decontamination methods will be employed to remove loose contamination within the contamination area. Best management practices for gross cleaning and/or decontamination of heavy equipment and vehicles consists of using wipers and nonhazardous materials to remove loose contamination.

One or more decontamination areas will be established either within the AOC or in the onsite area. Additional or final decontamination will be performed in these established areas. The amount of water used to clean equipment in the decontamination areas will be minimized. Water used will be raw or potable water only. Soaps, detergents, or other cleaning agents will not be added to the washwater. Pressure washing (if required) will normally be performed using cold water (hot water may be used to avoid icing). Steam cleaning will be used only after other decontamination methods prove to be ineffective.

Spent decontamination water and associated contamination will be discharged to the ground within the decontamination area. Verification sampling of the area will be performed in accordance with the project SAP prior to closeout of the project.

Decontamination practices will be documented in the daily field superintendent's log. Personnel responsible for equipment decontamination will be knowledgeable of the requirements of this section of the removal action work plan.

If cleanup of soils surrounding the buildings is too extensive to be managed in a cost-effective manner, the sites will then be stabilized in a manner that will not hinder future remediation. Future cleanup, if necessary, or transfer of the removal action scope will be coordinated with and approved by the lead regulatory agency. Future cleanup will occur at the same time that waste sites are addressed in the 100-FR-1, 100-DR-1 and 100-DR-2 Operable Units.

2.1.4 105-F Fuel Storage Basin

The fuel storage basin disposition plan (BHI 2000) was developed after various alternatives and removal methods were reviewed. The method chosen for basin cleanout combines safety and cost effectiveness using standard techniques, available equipment, and proven techniques. The disposition plan discusses the cleanout of the FSB (as part of the facility ISS) and the two stages of completion. Initially, a water-removal system will be fabricated, installed, and operated within the FSB. Water will be removed, tested, and disposed of at the 200 Area Effluent Treatment Facility. Stage I will address the FSB from the surface to approximately 5.2 m (17 ft) below grade. Stage II will address the remaining 0.9 m (3 ft) of basin material. Both phases will apply engineering controls to ensure worker safety, maintain radiological doses as low as reasonably achievable (ALARA), and minimize potential releases of radiological emissions to the environment. A readiness assessment will be conducted to assess work activity project readiness prior to initiating FSB cleanout work. A major portion of material/structure removal will use equipment (e.g., backhoes and excavators) that is typically used for D&D below-grade material removal, structure demolition, and large-scale soil remediation work. Experience gained in past basin cleanout projects (i.e., 105-B, 105-C, 105-D, 105-DR, and 105-N) will be used to optimize working conditions.

2.1.4.1 Upper Fill Removal (Stage I). Demolition of the above-grade structure will occur to allow access to the upper fill overburden, and characterization of the fill material. Excavation of backfill will begin at grade and continue to approximately -5.2 m (-17 ft) below grade, using primarily an excavator. The frequency of radiological surveys will increase as necessary as the removing backfill approaches the bottom of the basin. After being sampled and deemed appropriate for reuse, the upper fill material will be used as backfill upon EPA approval. The remaining soil layer will act as a biological shield and continue to confine the sediment and other radiological material located on or near the basin floor.

Originally, a retractable cover over the FSB was planned to be installed to provide weather protection . The need for such an enclosure was later reevaluated. Phase II excavation and demolition activities (Section 2.1.4.2) would be significantly easier and more efficient to perform without a cover and supporting structures. Unobstructed access to the basin allows for easier deployment of equipment. Supplemental lighting is minimized or eliminated. Subsequently, DOE and the lead regulatory agencies agreed that a cover would not be necessary (Smith 2000).

2.1.4.2 Debris and Contaminated Fill Removal with the Integrated Equipment System (Stage II). Removal of fuel elements or hot spot materials will be typically accomplished by using a small excavator (e.g., Brokk™-type remote control crawler) with attachments to expose the fuel elements or hot spot materials and then load the materials into an appropriate container. A large excavator or personnel with long-handled tools may be used to remove hot spot material

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and other lower fill debris and soil if dose rates do not pose a threat to the personnel. Loaded fuel containers will be placed into appropriate shipment containers prior to removal from the controlled area. Other hot spot materials will be excavated and packaged for disposal. Backfill material removed during Stage II will be radiologically screened, and any fuel elements will be remotely placed into a fuel holding container. Other hot spot material will be removed and packaged for disposal. Heavy equipment will provide shielding and distance for operators, allowing for prudent excavation of high-dose-rate items. (The identifying, locating, and removing of fuel elements and other hot spot materials are a first priority and are closely controlled by limitations outlined in the project authorization [safety] basis documents.)

The debris material (except for retrievable spent nuclear fuel) will be packaged for disposal at the ERDF or storage at the Central Waste Complex (CWC), as appropriate. Some waste materials may require segregation for macro-encapsulation, such as lead and/or high-dose materials (e.g., sediment). Typically, a loader track hoe, backhoe bucket, and attachments will hook, grip, shear, lift, or remove buckets, monorail beams, wood planking material, etc., and place the materials into ERDF-approved containers.

RESidual RADioactivity (RESRAD) dose model methodology may be used to determine radiological release levels for the basin wall, concrete floor, and the remaining structures, which is similar to the methodology used for the 105-C FSB and below-grade structures.

2.1.5 Safe Storage Enclosure Construction

Following decontamination and partial demolition of the reactor buildings, the existing shield walls will be used to create a SSE, including a new or enhanced metal roof. The shield walls will support the roof, and the enclosure will be completely sealed with only one entrance (i.e., a door will be welded shut). A utility room located outside of the SSE will be used for ventilation controls, monitoring equipment, and electrical power. Figure 2-1 shows a graphical representation of what the SSE will look like when completed.

After demolition activities have been completed, a new roof will be constructed over the remaining structure, and all penetrations and openings will be sealed. Exposed SSE concrete surfaces will be left "as-is" upon completion of ISS work.

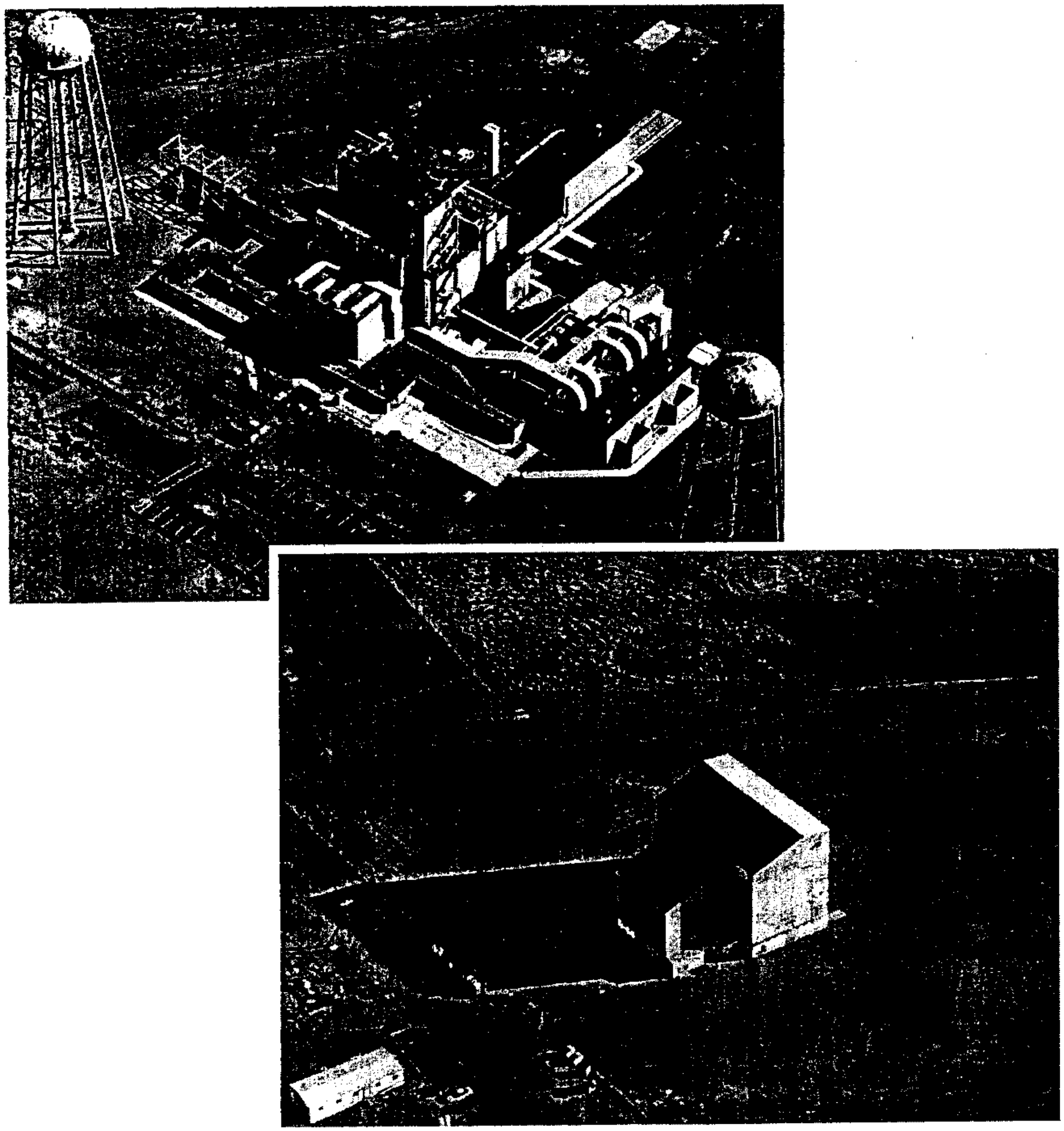
Key elements of the SSE design are as follows:

- Use existing shield walls as the basis for establishing the SSE footprint and as prime components of the structure.

- Remove floors and decking to within the minimal acceptable distance of the wall for integrity of the remaining structure.

- Close all wall penetrations securely so penetration closures will not be dislodged in a seismic event or from wind loads.

Figure 2-1. Safe Storage Enclosure.



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- Provide an access door for S&M activities.

- Provide for ventilation of the facility, if required, during surveillance inspections and maintenance activities.

- Address closure of all subsurface tunnels and pipes that will be left in place to prevent water or pest intrusion.

- Allow for decontamination of equipment and structural components to the extent reasonable for radioactive and hazardous material volume reduction, ALARA practices and, if practical, the release of material for unrestricted use.

- Provide a lighting system and convenience receptacles designed to provide adequate illumination for access/egress and adequate power for S&M activities. The SSEs will have a remote monitoring system to a central monitoring station for flooding within the facility and temperature monitoring.

2.1.6 Waste Disposal

All waste management activities will be performed in accordance with waste management ARARs identified in the action memorandum for the 105-F and 105-DR Reactor buildings and ancillary facilities (Ecology et al. 1998a). Waste from this action will either be disposed at the ERDF or at a lead regulatory agency and EPA-approved offsite disposal facility. Treatment of waste may be necessary prior to disposal at the ERDF, and waste may be stored at the ERDF with EPA concurrence while the waste is awaiting treatment.

If transuranic waste is encountered, storage is allowed at the Hanford Site's CWC on a case-by-case basis and requires lead regulatory agency and EPA approval. Certain material is eligible for salvage and recycling, which is encouraged if the appropriate regulatory requirements are met and it is economically feasible for the project to do so. Liquid waste will either be sent to the Hanford Site's Effluent Treatment Facility (ETF) or will be treated to meet the acceptance criteria of the receiving facility. Section 4.2 discusses waste management in further detail.

2.1.7 Site Restoration

Following the completion of demolition activities, if verification sampling of the site indicates that cleanup levels have been met, below-grade void spaces will be backfilled with nonhazardous/nonrecyclable material (e.g., clean concrete rubble and/or soil). Approximately the top 0.6 to 1 m (2 to 3.3 ft) will be backfilled with soil containing no greater than 20% cobble to facilitate future revegetation of the site. The final grade of the site will match the surrounding terrain. Existing barrow pits in the 100 Areas will be used. New barrow pits are not anticipated to be used for backfill.

If contamination is found in the soil during verification sampling, cleanup will proceed until it is determined to no longer be cost effective to continue. If this is the case, the site will be stabilized in a manner that will not hinder future remediation. The ISS Project final report (see Section 5.7) will provide information to identify the location (i.e., longitude, latitude, and depth) and the composition/concentration of contamination left behind. Determination of whether to proceed with soil cleanup and/or transfer of scope of work will be approved by the lead regulatory agency.

2.1.8 Demobilization

After verification sampling has been completed and the site has been graded to match the surrounding terrain, trailers, equipment, and personnel will be demobilized from the facility. Project closeout requirements are discussed in Sections 4.1 and 5.7.

2.2 FACILITY HAZARDS

The following section is a summary of information on facility hazards provided in the *Final Hazard Classification and Auditable Safety Analysis for the 105-DR Reactor Interim Safe Storage Project* (BHI 1998b) *Final Hazard Classification and Auditable Safety Analysis for the 105-F Building Interim Safe Storage Project* (BHI 1998c), and applicable Management of Change (MOC) documents. The MOC process is a DOE document change control process for a less than "Nuclear" facility used to determine if proposed changes or newly identified conditions require DOE notification and/or concurrence. As stated in Section 4.3, characterization of the ancillary facilities will be conducted to identify the potential existence of hazardous and radiological contamination.

2.2.1 Hazardous Material Inventory

The 105-F and 105-DR Reactor buildings have been deactivated and all bulk chemical inventories have been removed from the facilities for recycling or disposal. However, several types of hazardous materials remain in the facilities, including asbestos (in the form of cement asbestos board and insulation), mercury, lead, PCBs contained in light fixtures and paint, sodium dichromate, and equipment oil. Of these hazardous chemical constituents, asbestos and lead are found in the greatest quantities and are located throughout the facilities.

The potential for release of asbestos and lead is minimal because all asbestos and lead will be removed in accordance with approved procedures that ensure control over hazardous substances. Environmental Restoration Contractor (ERC) standards and procedures for asbestos and lead are intended to ensure that personnel control and handling and disposal of waste are performed in a manner that achieves the following objectives:

- Protects the safety of employees and the general public
- Minimizes spills and releases to the environment
- Meets applicable DOE, Federal, state, and local regulatory requirements.

Solvents, grease, oils (i.e., hydraulic oil and fuel), and aerosol containers have also been found throughout the facilities. Although the majority of these items were disposed during deactivation, the potential exists for personnel to find containers with residual chemical constituents. If such containers are found, the containers will be managed in accordance with Section 4.2.

2.2.2 Radiological Material Inventory

Radionuclide inventories may be found in all areas of the facilities, however, the only significant inventories are found in the reactor blocks and FSBs (BHI 1998b, 1998c). Criticality screenings and evaluations have been performed for the ISS activities for both facilities, which concluded that no potential exists for criticality (BHI 1998b, 1998c). These evaluations will be revised following completion of the ISS activities to reflect the fissile material inventory remaining within the SSE.

2.2.3 Facility Hazard Classification

A facility hazard classification, which is required for DOE facilities in accordance with DOE Order 5480.23, is based on an assessment of the potential release of hazardous and radiological inventories and their impacts to workers and the public. The results of the assessment of potential impacts are based on a bounding, unmitigated release of hazardous and radiological substances and a comparison to defined threshold values. The assessment of potential impacts considers the material quantity, form, location, dispersability, and interaction with available energy sources to determine unmitigated release potential. BHI-DE-01, *Design Engineering Procedures Manual*, Engineering Department Project Instruction (EDPI) 4.28-01, "Hazard Classification," establishes the basis for classifying a facility and the appropriate actions to be taken if a change in inventory is significant enough to change the facility classification or authorization basis.

The bounding scenario for the 105-F Reactor building is an unmitigated release from the FSB during a postulated unmitigated high-wind event (BHI 1998c). This release would result in a dose of 6.06 rem at 30 m (98.4 ft) to an individual (BHI 1998c), which is less than the derived threshold for Nuclear Category 3 of 10 rem, at 30 m (98.4 ft), in a 24-hour period (Bauer 1996). Therefore, the 105-F ISS Project is classified as radiological (Bauer 1996).

The bounding scenario for the 105-DR Reactor building is an unmitigated release from a postulated seismic event (BHI 1998b). This release would produce a dose of 6.74 rem at 30 m (98.4 ft) to an individual, which is also less than the derived threshold for Nuclear Category 3 (BHI 1998b). Therefore, the 105-DR Reactor building is also classified as radiological (Bauer 1996).

The auditable safety analyses for the 105-F and 105-DR Reactor buildings will be revised when the ISS projects are completed, consistent with the hazard and mission requirements for long-term S&M (up to 75 years).

2.3 STRUCTURES, SYSTEMS, AND COMPONENTS THAT PROTECT FACILITY WORKERS

Controls that will be employed during the 105-DR and 105-F ISS Projects include temporary confinement enclosures, glovebag containments, and personal protective equipment (PPE), as directed by the 105-DR or 105-F Reactor building health and safety plan (HASP). To control emissions, vacuums will be equipped with high-efficiency particulate air (HEPA) and/or charcoal filters, and exhausters will be equipped with HEPA filters. Radiological work permits (RWPs) and the asbestos abatement work plan for asbestos removal will also be in place. Personnel monitoring and area monitoring will be used as required to determine and document worker exposures and work conditions.

Temporary confinement enclosures will be constructed, as required, to provide proper air-flow conditions and will be fabricated of noncombustible and fire-retardant materials. One standard type of temporary confinement is glovebag enclosures. Glovebag enclosures will be essentially one-time-use protective measures used to prevent contamination release during specific operations (e.g., pipe cutting and sample collection). Glovebags are available in a variety of sizes and designs and will be ordered to tailored specifications in accordance with their intended uses.

2.3.1 Electrical System

2.3.1.1 105-DR Reactor Building. The electrical power to the 105-DR Reactor building will be deployed in three stages. The 105-DR electrical system was upgraded several years ago to provide a new power supply to meet current code requirements and to eliminate the original electrical system. This constituted the first stage. All of the existing plant power supply circuits were de-energized and deactivated prior to upgrading the power system to provide safety for future activities. The upgraded power supply was provided using a pole-mounted transformer to provide 120/240 volts, which is fed through a 600-amp/250-volt circuit breaker in storage room 1. This 120/240-volt power supply is fed through a new electrical system within the 105-DR Reactor building and provides power for lighting and outlet receptacles (this 120/240-volt system will provide power for early ISS project activities). However, power is insufficient for the full ISS project scope of work. Because this upgraded power system is internal to the 105-DR Reactor building, it will require deactivation later in the ISS project schedule. The next electrical stage will provide the power needed during the project to place the building into ISS.

The second electrical stage will provide full power for all ISS activities by installing the Mobile Integrated Temporary Utility System (MITUS), which first served at the 105-C ISS Project and will be deployed on the 105-DR ISS Project. The MITUS system provides an external power source to the 105-DR Reactor building, which allows internal power within the 105-DR Reactor building to be totally deactivated. The MITUS system consists of the following: a portable, trailer-mounted substation containing transformer and power distribution switchgear; load centers; and flexible temporary power cords connecting the substation and load centers. The load centers provide 120/240-volt ground-fault circuit interrupter outlet power and the load centers can be located in proximity to ISS project activities. The MITUS power supply will be independent of all internal 105-DR power, which will allow de-energization of the upgraded power supply stated above. This temporary MITUS will enhance safety during the ISS project because all permanent-wired power systems within the 105-DR Reactor building may be de-energized as directed by management. At the completion of the ISS project, the MITUS will be de-energized and made available for reassignment.

The last electrical stage will be installed during the SSE construction phase of the 105-DR ISS Project. The MITUS will then be replaced by a reduced-capacity, permanent-power system that will provide power for the ISS period of up to 75 years. The S&M activities have lower power demands than ISS project activities and require a reduced power supply system. The S&M power system will consist of a transformer providing 120/240 volts to lighting panels and outlets for remote monitoring instruments, lighting, and power for S&M activities. Any components existing from the initial two stages that can be used for the final stage of power will be considered for use.

2.3.1.2 105-F Reactor Building. The 105-F Reactor building's electrical system was upgraded several years ago to provide a new power supply to meet current code requirements. The upgraded electrical system eliminated the use of the original system, which was unsafe. All existing plant power supply circuits were de-energized and deactivated prior to upgrading the power system to provide safety for future activities. The upgraded power supply was provided by a pole-mounted transformer (120/240 volts). This 120/240-volt power is fed through a new electrical system within the 105-F Reactor building and provides power to lighting panels and outlets. The electrical power to the 105-F Reactor building will be deployed in three stages. The first phase will use the existing system, which will provide power for early ISS project activities; however, this power is insufficient for the full ISS project scope of work. Because the electrical system is internal to the 105-F Reactor building, it will require deactivation later during the ISS project schedule. The next electrical stage will provide the power needed during the project to place the building into ISS.

The second electrical stage will provide full power for all ISS activities by installing a mobile, temporary power system. The 105-DR Reactor building power system will have less capacity than MITUS and will not contain the paging/communication/alarm system or emergency lights of the MITUS. This temporary power system will consist of a substation with a transformer and power distribution switchgear, load centers, and flexible, temporary power cords connecting the substation and load centers. The load centers are portable and can be located in close proximity to ISS project activities. The temporary power system will be powered independently of all internal 105-F Reactor building power, which will allow de-energization of the upgraded power supply. This temporary power system will enhance safety during the ISS project because all permanent-wired power systems within the 105-F Reactor building may be de-energized as directed by management. At the completion of ISS, the mobile, temporary power system will be de-energized and made available for reassignment.

The third electrical stage will be installed during the SSE construction phase of the 105-F ISS Project. The mobile temporary power system will then be replaced by a reduced-capacity, permanent-power system that will provide power for the ISS period up to 75 years. The S&M activities have lower power demands than ISS project activities and require a reduced power supply system. The S&M power system will consist of a transformer providing 120/240 volts to lighting panels and outlets for remote monitoring instruments, lighting, and power for S&M activities. Any components existing from the initial two stages that can be used for the final stage of power will be considered for use.

3.0 SAFETY AND HEALTH MANAGEMENT AND CONTROLS

3.1 EMERGENCY MANAGEMENT

The ERC Emergency Management Program (including preparedness, planning, and response) is described in detail in BHI-SH-03, Volume 1, *Emergency Management Program*, and contains the administrative responsibilities for compliance with the *Hanford Emergency Response Plan* (DOE-RL 1998b). BHI-SH-03, Volume 2, contains emergency action plans for BHI-managed hazardous facilities. The 100-DR and 100-F Areas are combined into a single emergency action plan in BHI-SH-03, Volume 2, that identifies the capabilities necessary to respond to emergency conditions, provide guidance and instruction for initiating emergency response actions, and serve as a basis for training personnel in emergency actions for each facility. The emergency response actions within the emergency action plan are provided for recognizing incidents and/or abnormal conditions, initiating initial protective actions, and making the proper notifications. The emergency action plans are consistent with Hanford Site emergency procedures and meet the requirements of the *Hanford Emergency Response Plan* (DOE-RL 1998b), applicable DOE orders, and state and Federal regulations (i.e., *Code of Federal Regulations* [CFR] 29 CFR 1910.38 and WAC 173-303-340, -350, and -360).

All emergency planning and preparedness activities for these projects will be consistent with planning and preparedness actions taken by other Hanford Site contractors and similar projects. Activities will be in a manner that ensures the health and safety of workers and the public and the protection of the environment in the event of an abnormal incident at either the 105-DR or 105-F Reactor buildings or ancillary facilities (i.e., 116-D, 116-DR, 117-DR, and 119-DR).

Project response to any emergencies will be to evacuate personnel to a safe location and the required responsibilities of the Building Emergency Director and other project personnel who support the Incident Command System.

BHI-SH-03 (all volumes) comply with and implement the requirements of the *Hanford Emergency Response Plan* (DOE-RL 1998b) and applicable DOE orders. The Emergency Management Program establishes a coordinated emergency response organization capable of planning for, responding to, and recovering from industrial, security, or hazardous material incidents.

3.2 HEALTH AND SAFETY PROGRAM

3.2.1 Worker Safety Program

The ERC Hazardous Waste Operations Safety and Health Program was developed for employees involved in hazardous waste site activities. The program was developed to comply with the requirements of 29 CFR 1910.120 and 10 CFR 835 and to ensure the safety and health of workers during hazardous waste operations. The Integrated Environment, Safety, and Health

Management System will be incorporated into all work activities. The program includes the following elements:

- An organizational structure that specifies the official chain of command and the overall responsibilities of supervisors and employees
- A comprehensive work plan developed before work begins at a site to identify operations and objectives and to address the logistics and resources required to accomplish project goals
- A site-specific health and safety plan (SS HASP) where workers may be exposed to hazardous substances
- Worker training commensurate with individual job duties and work assignments
- A medical surveillance program administered to comply with the Occupational Safety and Health Administration (29 CFR 1910.120) requirements
- BHI-SH-02, *Safety and Health Procedures*, Volumes 1 through 4, and project/task-specific implementing plans and procedures
- Volunteer Protection Plan.

3.2.2 Site-Specific Health and Safety Plan and Activity Hazards Analysis

A SS HASP will be prepared that defines chemical, radiological, and physical hazards and specifies the controls and requirements for work activities. Building access and work activities are controlled by approved work packages, as required by established BHI/ERC procedures. The SS HASP addresses the health and safety hazards of each phase of site operation and includes the requirements of a SS HASP for hazardous waste operations and/or construction activities, as specified in 29 CFR 1910.120 and DOE Order 5480.9A. A SS HASP will be written for each ISS project and may include the ancillary buildings, however, a separate SS HASP may be developed for the ancillary buildings. As part of work package development, a job hazards analysis (JHA) will be written to identify the hazards associated with specific tasks not already covered under a SS HASP. The SS HASP elements include the following:

- Results of a risk and hazard analysis of each task and operation in the work plan
- List of employee training assignments
- List of PPE to be used by employees at the work site
- Medical surveillance requirements
- Work site control measures
- Emergency response
- Confined space entry procedures
- Spill containment program.

In addition to the SS HASP, a RWP will be prepared for work in areas with potential radiological hazards. The RWP extends the Radiological Protection Program (discussed in Section 3.2.3) to the specific work site or operation. All personnel assigned to the project and all work site visitors must strictly adhere to provisions identified in the SS HASP and RWP.

Before work and each activity begin, a pre-job briefing is held with the involved workers. This briefing includes reviews of the hazards that may be encountered and the associated requirements. Throughout an activity, daily briefings may also be held, as well as special briefings prior to major evolutions.

3.2.3 Radiological Controls and Protection

The Radiological Controls and Protection Program is defined in DOE-approved programs and BHI-approved procedures (BHI-RC-02 through BHI-RC-05, RadCon Procedure Manuals, and BHI-SH-01, *ERC Safety and Health Program*). This program implements the ERC's policy to reduce safety or health risks to levels that are ALARA and to ensure adequate protection of workers. The ERC Radiological Protection Program meets the requirements of 10 CFR 835. Radiological material handling will be managed in accordance with the DOE radiological control manual (DOE 1994). Appropriate dosimetry, RWPs, PPE, ALARA planning, periodic surveys, and radiological control technical support will be provided.

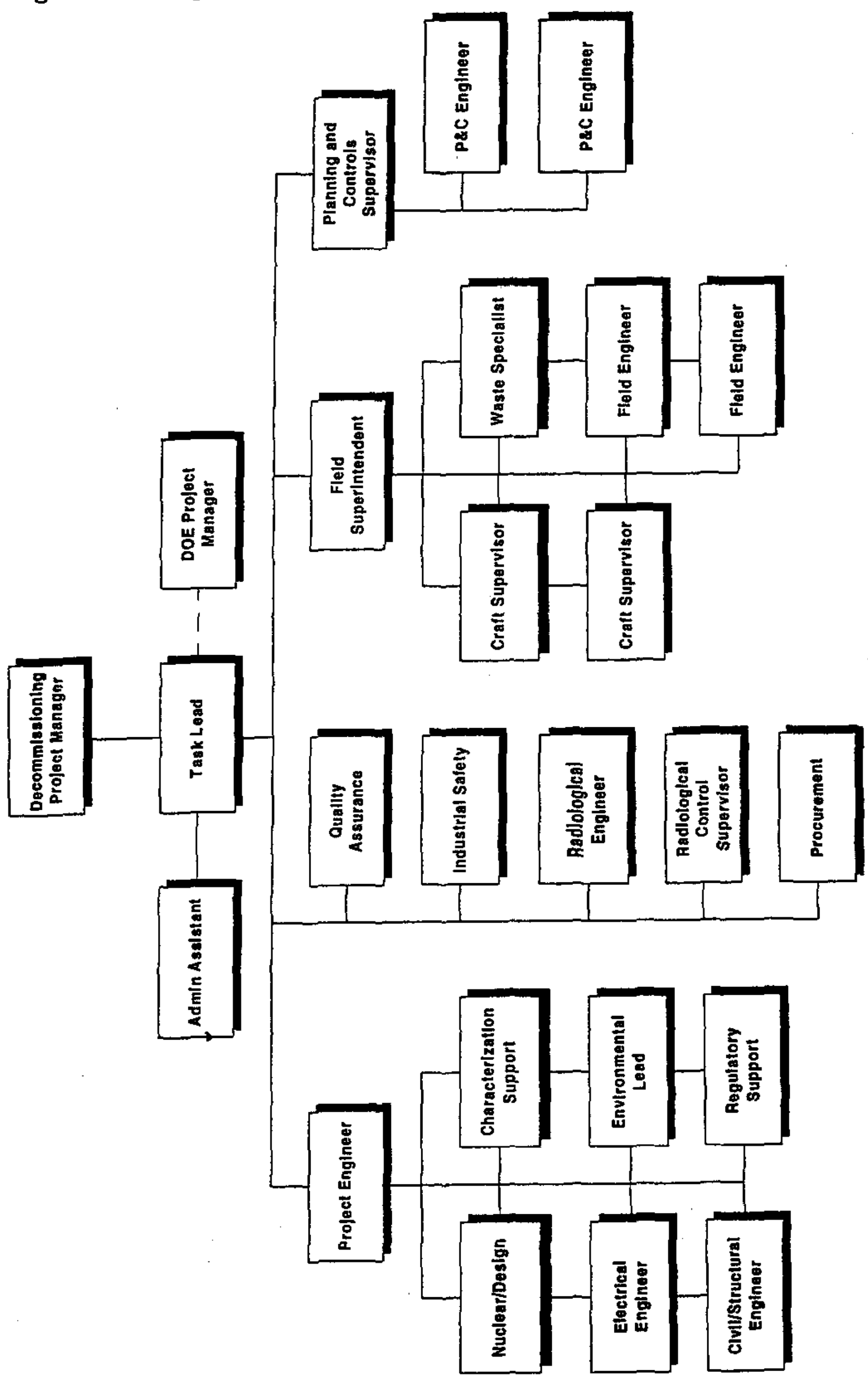
Standard ERC controls for work in radiological areas are assessed as adequate to control project activities. These controls provide for radiological controls planning that identifies the specific conditions and governs the specific requirements for an activity, periodic radiation and contamination surveys of the work area, and periodic or continuous observation of the work by the Radiological Control Organization. The ALARA planning process will identify shielding requirements, contamination control requirements (including local ventilation controls), radiation monitoring requirements, and other radiation control requirements for the individual tasks conducted during the course of the projects.

Measures are also taken to minimize the possibility of releases to the environment. Appendix B will quantitatively address the radionuclide inventory and activities that could cause potential release of this inventory, but not to the exclusion of the DOE radiological control manual (DOE 1994) or 10 CFR 835 requirements. Potential radiological air emissions are discussed in Appendix B.

3.3 MAINTENANCE MANAGEMENT

The BHI organizations responsible for the S&M and D&D programs have reached an agreement regarding the transition and responsibility of the facilities covered by this removal action scope. No systems will be maintained during decommissioning. Inspections, checks, and maintenance, which are activities normally conducted under S&M, will not be conducted during the D&D process. Access control to the respective facility during D&D activities will be under the control of the responsible field superintendent for the 105-DR or 105-F ISS Project (Figure 3-1).

Figure 3-1. Organization Chart for the 105-DR and 105-F ISS Projects.



4.0 ENVIRONMENTAL MANAGEMENT AND CONTROLS

4.1 APPLICABLE OR RELEVANT AND APPROPRIATE REQUIREMENTS

This section describes how each of the ARARs identified in the action memorandum (Ecology et al. 1998a) will be met, to the extent practicable, during the removal action.

- The *Model Toxics Control Act* (MTCA) (Method B), WAC 173-340, is applicable and specifies that cleanup actions must be protective of human health and the environment, comply with applicable state and Federal regulations to the extent practicable, and provide for compliance monitoring. The cleanup standards apply to soil, structures, and debris encountered during the removal action.

  Groundwater protection standards also apply if contaminated soil or structures remain in place. MTCA specifies that concentrations of residual contaminants are protective of groundwater at levels equal to or less than 100 times the MTCA groundwater cleanup levels, unless it can be demonstrated that a higher soil concentration is protective of groundwater at the site. If residual concentrations exceed cleanup levels calculated using the 100 times rule, site specific modeling will be performed to provide refinement on contaminants found to simulate actual site conditions.

  The MTCA cleanup standards for hazardous constituents in soil will be met for structures and/or demolition debris remaining after completion of the removal action. Remediation may be required for soil waste sites that fall within the layback area of the SSE. The MTCA cleanup levels will be met for soil waste sites remediated during the removal action.

  Achievement of cleanup standards will be verified through application of the MTCA cleanup levels and risk calculations Method B formula values for soil (Ecology 1996).

- RCRA, Subtitle C, is applicable regarding the generation, transportation, storage, treatment, and disposal of hazardous waste. Hazardous waste management regulations promulgated pursuant to RCRA are codified in 40 CFR 260 through 268. Regulations established under RCRA are applicable to any hazardous waste generated during the removal action. In addition, the State of Washington's dangerous waste regulations (WAC 173-303), which address the state program authorized under RCRA, are applicable for dangerous wastes encountered during the removal action. Additionally, this regulation applies for land disposal restricted waste, generator requirements, and transportation of hazardous wastes during the removal action.

  Disposition of hazardous substances from the facilities will be conducted in accordance with the waste generator requirements of RCRA, Subtitle C (40 CFR 262), and WAC 173-303, including waste designation, waste storage prior to disposal, and disposal restrictions. Waste disposal will also be governed by the requirements of the RCRA land disposal restriction

(40 CFR 268); radiological waste land disposal requirements of 10 CFR 61, Subpart C; and the ERDF waste acceptance criteria (BHI 1998d) for onsite disposal.

- The *Toxic Substances Control Act of 1976* (TSCA) (40 CFR 761) regulates the management and disposal of PCBs and PCB waste. All waste suspected to contain PCBs will be evaluated to determine if the waste meets ERDF waste acceptance criteria (BHI 1998d). Any PCB waste that does not meet ERDF waste acceptance criteria will be disposed offsite at a lead regulatory agency and EPA-approved facility capable of accepting TSCA waste.

- "U.S. Department of Transportation Requirements for the Transportation of Hazardous Materials" (49 CFR 100-179) are applicable for any wastes transported from the Hanford Site.

- The *Hazardous Materials Transportation Act of 1974* is applicable for transportation of potentially hazardous materials, including samples and waste. All offsite shipments for disposal will comply with applicable packaging, marking, labeling, and shipping requirements. Any shipment of potentially hazardous materials, either onsite or offsite, will also comply with *the Hazardous Materials Transportation Act.*

- The *Clean Air Act of 1955* (40 CFR 61) is applicable to releases of airborne contaminants that may occur during the removal action, as well as the air monitoring requirements for these contaminants. Specifically, Subpart H provides the standards to ensure that emissions from radionuclides are minimized during collection, processing, packaging, and transportation. These standards are applicable to radionuclides that may be encountered during the removal action to prevent exceeding 10 mrem/yr effective dose equivalent to any member of the public. Subpart M and the Occupational Safety and Health Administration (29 CFR 1910.1101 and WAC 296-62) define the regulations pertaining to removal and disposal of asbestos and ACM and provide special precautions to prevent exposure of workers to airborne emissions of asbestos fibers. Compliance with these regulations during the removal action will satisfy the requirements of these ARARs.

- "Radiation Protection—Air Emissions" (WAC 246-247) is applicable to the release of airborne radionuclides that may occur during the removal action, as well as the air monitoring requirements and best available radionuclide control technology.

  Washington State Department of Health regulations govern the release of airborne radionuclides (WAC 246-247). Quantifying radioactive emissions, implementing best available radionuclide control technology, and performing air monitoring for emission verification have been identified as substantive requirements. Appendix B provides detailed information demonstrating how these requirements will be met. The calculated unabated offsite dose, 7.55×10^{-4} mrem/yr for the 105-DR Reactor building (which includes the 116-D and 116-DR stacks), and 9.80×10^{-3} mrem/yr for the 105-F Reactor building (including FSB activities) are not greater than 0.1 mrem/yr; therefore, this activity is not subject to the substantive requirements of 40 CFR 61, Subpart H (i.e., National Emission Standards for

Hazardous Air Pollutants compliant monitoring). However, periodic confirmatory measurements, as described in Appendix B, will be conducted.

- "General Regulation for Air Pollution Sources" (WAC 173-400) and "Controls for New Sources of Toxic Air Pollutants" (WAC 173-460). Standard demolition and dust control techniques employed during ISS provide reasonable precautions to control fugitive emissions. These techniques are the same as those applied to control radionuclide air emissions as described in Appendix B. These techniques include, but are not limited to, the application of water and fixatives, temporary confinement enclosures/glovebag containments which may be HEPA filtered, and HEPA filtered and/or charcoal filtered vacuums.

- The *Safe Drinking Water Act of 1974* and "National Primary Drinking Water Regulations" (40 CFR 141, Subpart B) for public drinking water supplies establish cleanup goals that are protective of groundwater. Although the removal action does not directly address groundwater cleanup in the 100-D/DR and 100-F Areas, below-grade structures, soil, and demolition debris to be left in place will be remediated to meet standards that are protective of groundwater. Protectiveness will be verified by ensuring that soil cleanup levels for materials left in place allow compliance with maximum contaminant levels (MCLs) for hazardous and radioactive constituents in groundwater. The RESRAD dose models will be used to verify protectiveness with regard to radiological constituents, as described in Section 4.4. Soil not addressed by this removal action in the vicinity of the facilities that may be contaminated will be addressed in the final remedial action for the 100-DR-2, 100-DR-1, and 100-FR-1 Operable Units.

- The *Clean Water Act of 1977*, as implemented by WAC 173-201A, establishes cleanup goals that address protection of the Columbia River. Erosion and stormwater controls will be used as necessary during and following the removal action to prevent wastewater/stormwater discharges directly to the Columbia River. Building material, soil, and demolition debris to be left in place will meet standards that are protective of the Columbia River.

- The *National Historic Preservation Act of 1966* (implemented through 36 CFR 800) requires Federal agencies to evaluate and mitigate adverse effects of Federal activities on any site eligible for inclusion on the National Register of Historic Places. The programmatic agreement for the maintenance, deactivation, alteration, and demolition of the built environment allows DOE to prepare a treatment plan that provides for the mitigation of historic structures. The programmatic agreement requires that all mitigation activities identified in the treatment plan be completed before demolition, alteration, or removal of artifacts. Although the 105-DR and 105-F Reactor buildings were determined eligible for listing on the National Register of Historic Places, the facilities were not recommended for mitigation (Neitzel 1997), and no further action is required to meet this ARAR.

- The *Archeological Resources Protection Act of 1979* (43 CFR 37) would govern the protection of any significant artifacts that may be found during the removal action. Because of the extensive disturbance resulting from construction of the facilities, it is unlikely that archaeological remains will be found in the footprint of the facilities (Neitzel 1997).

However, if archeological remains are discovered, a mitigation plan will be developed in consultation with the appropriate authorities. Section 4.5, "Natural and Cultural Resources Protection," discusses this subject in more detail.

- The *Endangered Species Act of 1973* (implementing regulations of 50 CFR 402) and WAC 232-012-297 prohibit activities that threaten the continued existence of listed species or that destroy critical habitat. Threatened and endangered species are known to be present in the 100 Areas, but no adverse impacts on protected species or critical habitat are anticipated from activities associated with the removal action. An ecological review will be conducted prior to demolition to identify any potential impacts. If potential impacts are discovered, an appropriate mitigation plan will be developed and implemented.

- The *Native American Graves Protection and Repatriation Act* (implemented via 40 CFR 10) requires agencies to consult with and notify culturally affiliated tribes when Native American remains are inadvertently discovered during project activities. It is unlikely that the removal action would inadvertently uncover human remains. If human remains are encountered, pre-established procedures, documented in the *Hanford Cultural Resource Management Plan* (PNL 1989), will be followed.

4.1.1 Other Criteria, Advisories, or Guidance to be Considered for this Removal Action

In addition to the ARARs identified in the action memorandum and discussed in Section 4.1 the following criteria, advisories, and guidance will be complied with in accordance to the action memorandum (Ecology et al. 1998a) during implementation of the removal action. These materials, while not promulgated as regulations, are important to protect human health and the environment and to protect workers during the implementation phase.

- *Establishment of Cleanup Levels for CERCLA Sites with Radioactive Contamination* (EPA 1997a) is an EPA policy statement that provides clarifying guidance for establishing cleanup levels for radioactive contamination at CERCLA sites. The statement provides guidance regarding protection of human health and does not address cleanup levels necessary to protect ecological receptors. It should be noted, however, that for most radionuclides, remediation goals that are protective of human health are also considered protective of ecological receptors. The guidance indicates that cleanup levels should consider exposure from all pathways and through all media (e.g., soil, groundwater, surface water, sediment, air, structures, and biota). The policy statement establishes a human health risk range of 10^{-4} to 10^{-6}, which is roughly equivalent to 15 mrem/yr effective dose equivalent as the maximum dose limit for humans. It further states that background should be determined on a site-specific basis. Although not an ARAR, the cleanup standard in the EPA policy statement must be addressed to satisfy the threshold criterion for protectiveness.

  Consistent with the risk range, EPA has considered cancer risk from radiation in a number of different contexts and has concluded that levels of 15 mrem/yr above background are protective of human health and the environment. Additionally, the impact to groundwater may not exceed the MCLs . In general, below-grade structures will be removed to a minimum of 0.9 m (3 ft) below surrounding grade. However, if any of the cleanup factors cannot be met, the portions of the below-grade structures and soils above cleanup levels will be removed. These waste sites will meet the rural-residential cleanup scenario, as previously agreed to in the *Remedial Design Report/Remedial Action Work Plan for the 100 Area* (DOE-RL 1998c). In the event that large volumes of contaminated soil are encountered or removal of contaminated soil inhibits reactor safe storage activities, with concurrence by the lead regulatory agency, the removal of contaminated soils may be deferred to the RA/WD Program (Ecology et al. 1998a).

- The ERDF waste acceptance criteria (BHI 1998d) and *Supplemental Waste Acceptance Criteria for Bulk Shipments to the Environmental Restoration Disposal Facility* (BHI 1997b) delineate primary requirements including regulatory requirements, specific isotopic constituents and contamination levels, dangerous/hazardous constituents and concentrations, and physical/chemical waste characteristics that are acceptable for disposal of wastes at the ERDF. Prior to disposal, waste will be evaluated to ensure that the waste meets ERDF waste acceptance criteria.

  Revised Procedures for Planning and Implementing Off-Site Response Actions (EPA 1987) provides procedures for offsite disposal of CERCLA wastes. The policy was explicitly superceded in September 1993 (58 FR 49200) with the promulgation of EPA's CERCLA Off-Site Rule, which established the requirements for off-site actions as a regulatory requirement within 40 CFR 300.440. Although it is anticipated that waste generated by the removal action will be disposed onsite, these procedures will be implemented for any offsite disposal that would be required. The EPA Remedial Project Manager will be responsible for decisions regarding the offsite disposal of regulated waste generated during the removal action.

- *Hanford Site Solid Waste Acceptance Criteria* (FDH 1998) identifies the criteria for acceptance of waste at the CWC and ETF.

- "Radiation Protection Guidance for Exposure to the General Public" (59 FR 66414) provides EPA protection guidance recommending that nonmedical radiation doses to the public from all sources and pathways not exceed 100 mrem/yr above background. It also recommends that lower dose limits be applied to individual sources and pathways. One such individual source is residual environmental radiation contamination after the cleanup of a site. The removal action will meet a 15 mrem/yr effective dose equivalent goal, excluding nearby waste sites to be addressed by the remedial action program.

- "Occupational Radiation Protection" (10 CFR 835) establishes radiation protection standards, limits, and program requirements for protecting workers from ionizing radiation resulting from the conduct of DOE activities. It also requires that measures are taken to maintain radiation exposures ALARA. A combination of PPE, personnel training, physical design features (e.g., confinement, remote handling, and shielded containers), and administrative controls (e.g., limiting time in radiation zones) would be used to ensure that the requirements for worker and visitor protection are met. In addition, the requirements to maintain exposure ALARA will be achieved by decontaminating surfaces to the extent practicable prior to demolition and by providing PPE, training, and administrative controls. For surfaces that could not be adequately decontaminated, fixatives would be applied to contaminants to ensure exposure ALARA. Individual monitoring would be performed as necessary to verify compliance with the requirements.

- Exposure limits, personnel protection requirements, and decontamination methods for hazardous chemicals are established by 29 CFR 1910. The regulation also requires identification and mitigation of physical hazards to workers posed by a facility including, but not limited to, confined spaces, falling hazards, fire, and electrical shock. The regulation provides requirements for worker safety during construction activities.

- *Radioactive Waste Management* (DOE Order 5820.2A) provides the requirements for managing low-level radioactive waste and transuranic waste.

- It is the DOE's interpretation that the *National Environmental Policy Act of 1969* (NEPA) requires CERCLA documents to address values of NEPA (DOE 1994b). The EE/CA (DOE-RL 1998a), which is a CERCLA document, incorporated NEPA values to the extent practicable.

- The Tri-Party Agreement (Ecology et al. 1998b) contains provisions governing RCRA and CERCLA cleanup activities at the Hanford Site and provides guidance on integrating RCRA and CERCLA requirements to the greatest extent practicable. These provisions are applicable during this removal action, and the requirements have been identified and addressed through the ARARs, as well as documented in the project schedule.

4.2 WASTE MANAGEMENT

CERCLA Section 104(d)(4) states that where two or more noncontiguous facilities are reasonably related on the basis of geography, or on the basis of the threat or potential threat to the public health or welfare or the environment, the President may, at their discretion, treat these facilities as one for the purposes of this section. The preamble to the National Contingency Plan (40 CFR 300) clarifies the stated EPA interpretation that when noncontiguous facilities are reasonably close to one another and wastes at these sites are compatible for a selected treatment or disposal approach, CERCLA Section 104(d)(4) allows the lead agency to treat these related facilities as one site for response purposes and, therefore, allows the lead agency to manage waste transferred between such noncontiguous facilities without having to obtain a permit. Therefore, the facilities in the 100 Areas addressed by this removal action work plan and the various disposal/storage facilities such as the ERDF, CWC, and ETF, which are in the 200 Areas, are considered as a single site for response purposes under this removal action work plan.

Waste management activities will be performed in accordance with waste management ARARs identified in the action memorandum for the 105-DR and 105-F Reactor buildings and ancillary facilities (Ecology et al. 1998a) and as discussed above in Section 4.1. The requirements specified by the ARARs and other applicable guidance will be addressed in a site-specific waste management instruction prepared in accordance with BHI-EE-10, *Waste Management Plan*, Part II, Procedure 5.0 , "Site-Specific Waste Management Instructions." The site-specific waste management instruction will address waste storage, transportation, packaging, handling, and labeling as they specifically apply to waste streams.

In conducting the removal action, various waste steams will be generated. Each waste stream will require specific processing and disposal. These waste streams will include the following:

- Solid waste
- Low-level radioactive waste
- Mixed waste (waste that is both low-level radioactive waste and hazardous waste)
- Used oil
- Hazardous, dangerous, and PCB wastes
- Transuranic waste.

4.2.1 Waste Characterization and Designation

Waste generated will be characterized and designated in accordance with BHI-EE-10, *Waste Management Plan*; the requirements of the receiving facility; and the approved SAP. Waste will be segregated by radioactive content, physical form, and chemical form. The generation of waste will be minimized to the maximum extent practical. Wastes destined for the ERDF will be:

- Characterized in accordance with the appropriate project SAP
- Designated in accordance with the following:
  - BHI-EE-10, Part I, Procedure 2.0, "Waste Identification" ; Part II, Procedure 2.0, "Waste Certification"
  - ERDF waste acceptance criteria (BHI 1998d)
  - State of Washington Dangerous Waste Regulations (WAC 173-303).

Waste destined for one of the Project Hanford Management Contractor (PHMC)-controlled facilities will be designated and characterized in accordance with BHI-EE-10, *Hanford Site Solid Waste Acceptance Criteria* (FDH 1998), and the approved SAP.

4.2.2 Waste Handling, Storage, and Packaging

In general, disposal of waste generated from the removal actions discussed in this work plan will either be disposed at the ERDF or at a lead regulatory agency and EPA-approved offsite disposal facility. Treatment of waste may be necessary prior to disposal at the ERDF. If transuranic waste is encountered, storage will be allowed at the Hanford Site's CWC on a case-by-case basis and requires lead regulatory agency and EPA approval. Liquid waste shall either be sent to the Hanford Site's ETF, shipped offsite to a lead regulatory agency and EPA-approved facility , or re-used for dust suppression with lead regulatory agency approval. Ecology approval will be obtained prior to disposal of any waste streams sent to the ETF. Specific details on waste handling, storage, and packaging for the variety of wastes that may be encountered during the removal actions are discussed below.

Waste minimization practices will be followed to the extent technically and economically feasible during all phases of waste management. Waste materials will be recycled, reused, or reclaimed when feasible. Introduction of clean materials into a contamination area and contamination of clean materials will be minimized to the extent practicable. During all phases of waste management, emphasis will be placed on source reduction to eliminate or minimize the volume of wastes that will be generated.

Asbestos will be adequately wetted and double-bagged or double-wrapped in plastic, according to the field-specific asbestos abatement work plan, which is maintained at the site. Packages will be limited to 18.2 kg (40 lb). Cement asbestos board has no weight restriction per package. Cut and wrapped pipe will be packaged to meet the requirements of the waste shipping and receiving plan for asbestos on pipe.

Biological wastes will be packaged in strong-tight containers that will not leak during storage. Generally, liquids will be collected in 209-L (55-gal) UN1A2 drums. However, the size of the container (e.g., 57-L, 114-L, and 209-L [15-gal, 30-gal, and 55-gal]) may vary depending on the volume of material to be packaged. Aqueous solutions with a pH > 2 and < 12.5 may be stored in a large storage tank or in drums.

Aqueous solutions with a pH $\leq$ 2 and $\geq$ 12.5 will be stored in 209-L (55-gal) UN1A2 drums. Signs stating, "DANGER-UNAUTHORIZED PERSONNEL KEEP OUT," will be posted at each entrance of the storage area and along the boundary as necessary to be seen from any approach to the area. Portable fire extinguishers and spill-control equipment will be available. Containers will not be opened, handled, or stored in a manner that may rupture the container or cause the container to leak. Containers in poor condition will have the contents transferred to a container in good condition. A minimum 76.2-cm (30-in.) separation will be maintained between container rows. A row of containers will be no more than two containers wide.

Aqueous solutions that are stored in the large holding tanks will be shipped by tanker truck to the ETF upon Ecology approval. Nonbulk containers of aqueous solutions that cannot be combined in the large holding tank due to incompatibility or the presence of hazardous characteristics that cause the solutions to designate as hazardous, nonradioactive waste may be treated in accordance with the approved waste treatment plan and sent to the ERDF. If waste is encountered for which there is no available treatment, DOE will meet with the regulatory agencies to determine the appropriate action for the waste stream. The preference for treatment of large-volume waste streams is the ETF. The ERDF will only be used when small volumes of waste needing treatment are encountered. Mixed (hazardous and radioactive) liquids may be treated and shipped to the ERDF. Nonradioactive, nonhazardous liquids will be shipped, reused, or recycled. Hazardous liquids (nonradioactive) will be treated and shipped to the ERDF or, with lead regulatory agency and EPA approval, shipped offsite for disposal.

Smaller items contaminated with mixed (radioactive and dangerous/hazardous) solids will be packaged in 209-L (55-gal) drums (UN1A2). The weight will not exceed 385.9 kg (850 lb). Larger pieces (e.g., bricks and sheets) shall be double-wrapped in plastic and palletized.

High-dose radioactive items will be placed in U.S. Department of Transportation (DOT)-approved containers and shipped to a facility that is approved by the lead regulatory agency and EPA. For onsite shipments, high-dose radioactive items will be placed in containers specified by the receiving facility.

Demolition concrete will be reduced to approximately 0.9 m^2 (1 ft^2). Loose rebar will be reduced to lengths of approximately 1.2 m (4 ft). Size reduction prevents materials from becoming lodged and/or damaging the ERDF containers. Structural pieces will be sized to fit to ensure that ERDF acceptance criteria (BHI 1998d) are met.

Demolition debris coated with paints that contain PCBs $\geq$ 50 ppm meet the TSCA definition of "PCB bulk product waste." The debris will be managed within the AOC until such time it is loaded into an ERDF container in the onsite area. The ERDF container will be marked with a M_L marking (CAUTION-CONTAINS PCBs) and will be closed and securely fitted with a tarp except when adding or removing waste.

All containers, packages, or items requiring storage in a radioactive material area will be marked/labeled with radioactive material markings and unique consecutive identification numbers. Containers or packages of waste requiring tracking (e.g., hazardous, mixed) will be assigned a package identification number by a Waste Transportation Specialist.

Nonradioactive solid items will be packaged in 209-L (55-gal) drums (UN1A2). Larger items will be double-wrapped in plastic and palletized. Radioactive solids will be placed in bulk roll-off containers with side-swinging gates (400 and 700 series) for ERDF disposal. The containers will be lined with plastic sheeting and covered by a tarp. Lightweight material such as paper and plastic will be bagged prior to placing in the container to eliminate the potential of materials blowing out of the container.

Nonradioactive solids that designate as dangerous and do not meet ERDF waste acceptance criteria may, with lead regulatory agency and EPA approval, be shipped as dangerous waste offsite or to the 1100 Area Excess Yard or the 400 Area Consolidation Center if the material is recyclable. Lead regulatory agency and EPA approval will be obtained to ship PCB (TSCA) waste to offsite TSCA disposal facilities.

Areas containing PCB oils will be marked with signs posting, "DANGER-UNAUTHORIZED PERSONNEL KEEP OUT," at each entrance and along the boundary as necessary to be seen from any approach to the area. The M_L marking will also be posted. Portable fire extinguishers and spill-control equipment will be available. Containers will not be opened, handled, or stored

in a manner that may rupture the container or cause the container to leak. Containers in poor condition will have the contents transferred to a container in good condition. A minimum 76.2-cm (30-in.) separation will be maintained between container rows. A row of containers will be no more than two containers wide.
Storage of all containers (except for containers used to collect fluorescent light tubes) will be closed and secured when not being filled or emptied. Radioactively contaminated waste will be stored in a radioactive materials area that is established, managed and maintained in accordance with established BHI procedures. Containers will be stored to prevent the accumulation of water.

The *onsite area* includes the areal extent of contamination as well as all suitable areas in very close proximity to the contamination necessary for implementation of the removal action (40 CFR 300.400(e)). The AOC consists of the area of continuous contamination of various amounts and types (Shapiro 1996). This includes the footprint of the building itself and the adjacent areas where generally dispersed contamination may reside due to past–practice activities. Figures 4-1 and 4-2 show the locations of the AOC and the onsite area. They also show waste ~~staging~~ storage areas and bulk ERDF container transportation queue areas located within the onsite areas that will be used at the 105-F and 105-DR Areas.

4.2.3 Waste Treatment

Treatment of waste streams may be necessary to provide for safe transport or effective disposal. The type of treatment and the location of treatment will be determined by DOE and the lead regulatory agency on a case-by-case basis, in accordance with the substantive requirements of RCRA and WAC 173-303. Upon lead regulatory agency approval, solidification, encapsulation, and neutralization may be employed to treat various wastes. Other treatment methods may be considered if necessary. For wastes requiring treatment, the techniques will be documented in site-specific waste management instructions or equivalent treatment document, which will be approved by the lead regulatory agency.

4.2.4 Waste Transportation and Shipping

When transportation subcontractor services are used for waste generated during the 105-DR and 105-F Reactor buildings removal action, the subcontractor will be responsible for using and maintaining appropriate transport motor vehicles and providing qualified commercial drivers. All shipments will be made in accordance with DOT regulations, 49 CFR 171-179, applicable sections of WAC 173-303, and BHI-EE-12.

4.2.5 Disposal

Contaminated soil encountered during demolition of the facilities and soil in the FSBs will be disposed in an appropriate disposal facility. The remaining contaminated soil waste sites will be addressed by DOE's remedial action program, as specified in other CERCLA decision documents.

Figure 4-1. 105-F Waste Staging Areas.

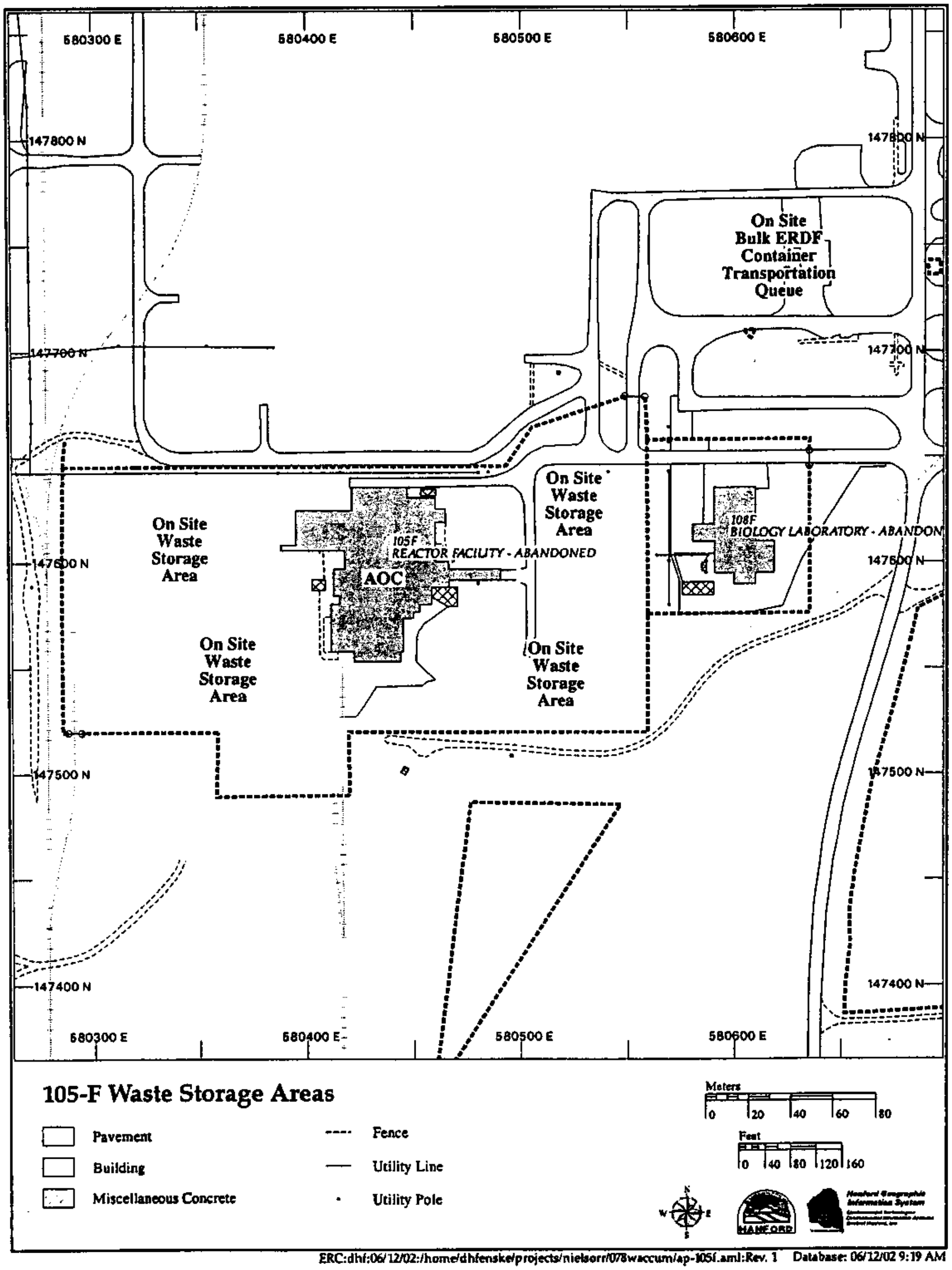
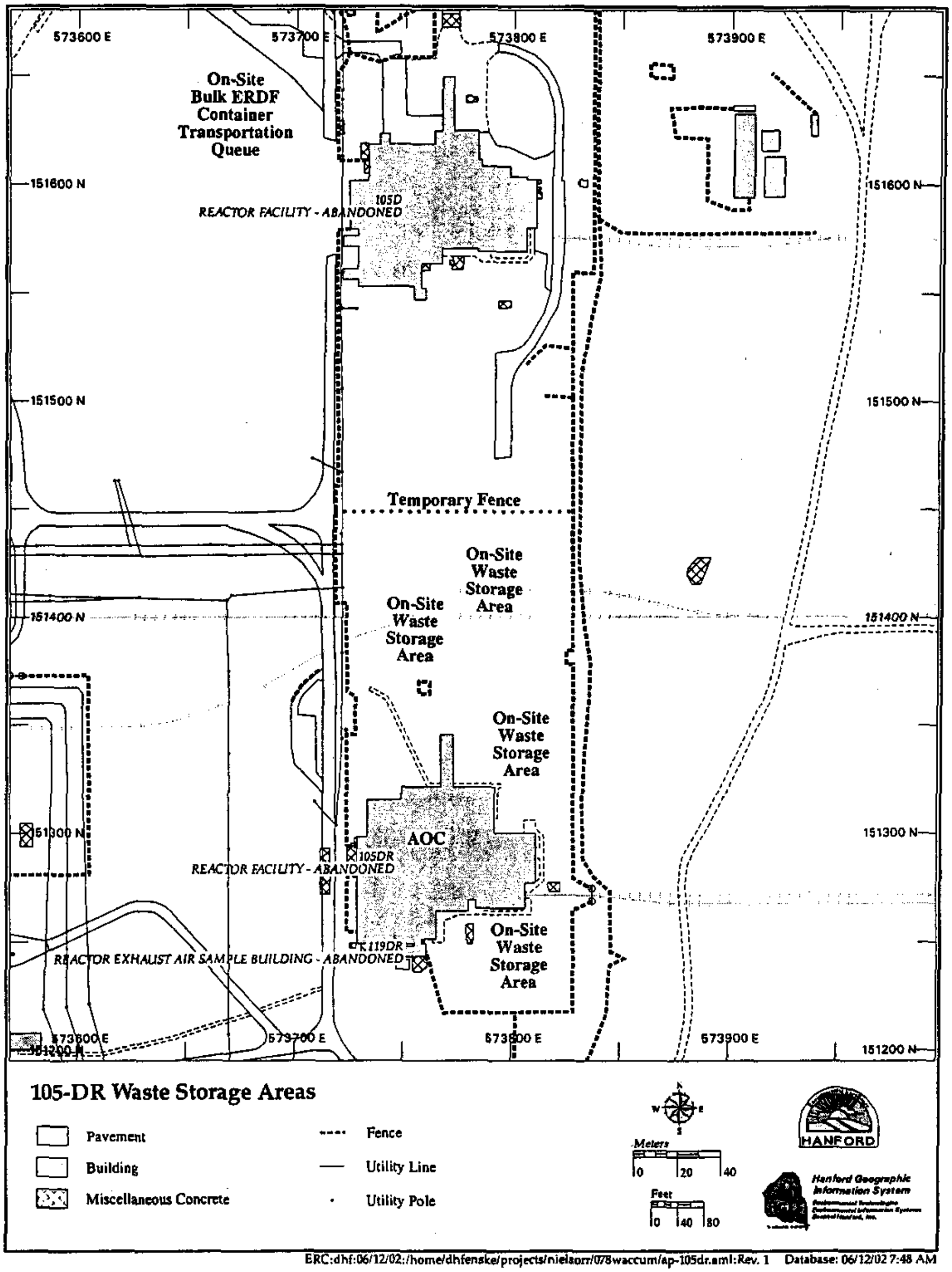


Figure 4-2. 105-DR Waste Staging Areas.



Waste resulting from this action will either be disposed at the ERDF or a lead regulatory agency and EPA-approved offsite disposal facility. Treatment of waste may be necessary prior to disposal at ERDF. Should transuranic waste be encountered, storage will be allowed at the Hanford Site's CWC and requires lead regulatory agency and EPA approval. Certain material is eligible for salvage and recycling, which is encouraged, provided that the appropriate regulatory requirements are met and that it is economically feasible for the project to do so. In addition, materials shipped offsite for salvage or recycle must be certified free of radioactive contamination in accordance with the ERC's material release program, as discussed in Section 4.4. Liquid waste will either be sent to the Hanford Site's ETF or treated to meet the acceptance criteria of the receiving facility. Ecology approval is required prior to shipping contaminated water to the ETF for treatment. *Hanford Site Solid Waste Acceptance Criteria* (FDH 1998) identifies criteria for acceptance of waste at the CWC and ETF. *Environmental Restoration Disposal Facility Waste Acceptance Criteria* (BHI 1998d) and Supplemental Waste Acceptance Criteria (BHI 1997b) provide the waste acceptance criteria for the ERDF.

Small volumes of uncontaminated concrete rinse water and dilute concrete slurry may be generated from rinsing concrete trucks and pumping units. These small volumes will be discharged to soils immediately adjacent to the reactors in areas that have met interim closure requirements of the applicable ~~sampling and analysis plan~~SAP. Discharges will occur in a variety of locations rather than in a single location. Water used for rinsing purposes will either be raw or potable water. Reasonable efforts will be taken to minimize volumes and prevent ponding and runoff.

4.2.6 Waste Management Strategy

Basic waste management strategies are discussed below; however, if more cost-effective methods become available, other methods may be used. Throughout the project, material will be recycled whenever possible, assuming that it is economically feasible for the project.

- **Solid waste:** Solid waste will be managed in accordance with WAC 173-304, with an emphasis on recycling or reuse to the maximum extent possible. Solid waste will primarily be sent to inert demolition waste landfills and for offsite disposal at a municipal/industrial landfill (e.g., the City of Richland landfill). All materials released offsite for disposal, recycle, or salvage must be certified free of radioactive contamination in accordance with the ERC's material release program, as discussed in Section 4.4.

- **Low-level radioactive waste:** Low-level radioactive waste that meets ERDF waste acceptance criteria (BHI 1998d) will be disposed at the ERDF.

- **Mixed waste:** RCRA mixed waste will be managed in compliance with the requirements for both hazardous/dangerous wastes (WAC 173-303) and radioactive waste (10 CFR 61). Where treatment is deemed not feasible, approval will be obtained from the lead regulatory agency and EPA to ship waste to an approved TSD facility. If mixed waste streams are found in quantities large enough to make treatment a viable option, mixed wastes may be treated to meet applicable land disposal restrictions and disposed at the ERDF. Small volumes of waste may be treated or accumulated for later treatment.

- **Used oil:** Most used oil identified is nonradioactively contaminated. The preferred strategy is to manage the oil (except for PCB oils) as a recyclable material in accordance with the Hanford Site-wide used oil program. Used oil will be evaluated with the material release program, as discussed in Section 4.4.

- **Hazardous/dangerous wastes:** Hazardous/dangerous wastes in the facilities consist primarily of mercury, lead, sodium dichromate, and PCBs. Some forms of mercury can be treated as a recyclable material (if not radioactive). If any of these wastes are found to be radioactive, they will be treated as mixed waste. Waste that cannot be treated to meet the ERDF waste acceptance criteria may be shipped to an offsite TSD facility, contingent upon the waste meeting the offsite disposal facility's waste acceptance criteria and obtaining an offsite determination of acceptability by the lead regulatory agency and EPA. In addition, waste shipped offsite for disposal will be certified free of radioactive contamination in accordance with the ERC's material release program, as discussed in Section 4.4.

- **Transuranic wastes:** Without regard to source or form, waste that is contaminated with alpha-emitting transuranic radionuclides with half-lives greater than 20 years and concentrations greater than 100 nCi/g at the time of assay is classified as transuranic waste (DOE Order 5820.2A). Transuranic waste will be managed in accordance with BHI-EE-10. The CWC, operated by the PHMC, will be used for interim storage of any transuranic waste encountered. Storage of transuranic waste at the CWC requires lead regulatory agency and EPA approval.

- **Accountable nuclear materials:** If accountable nuclear materials (e.g., spent nuclear fuel elements or pieces) are discovered in the sediment, the materials will be packaged and stored in accordance with Hanford Site security requirements. (Potential storage locations for onsite storage are identified in Figures 4-1 and 4-2.) With concurrence of the lead regulatory agency, this material will be transferred to the PHMC for disposal in accordance with a memorandum of understanding.

4.3 CHARACTERIZATION

Characterization (through sampling and analysis and radiological surveys) will be conducted throughout the 105-DR and 105-F ISS Projects to identify radiological and hazardous conditions that will be encountered during facility operations. These technical services will also be used to identify and characterize waste streams for treatment and/or disposal and to verify facility and area conditions at various phases of project completion. They will also determine if the appropriate cleanup standards have been met. Analytical data generated in these efforts will be used to develop the following information:

- Contaminant identification
- Contaminant concentrations
- Compliance with cleanup standards
- Waste type categories
- Worker health and safety conditions
- Decontamination requirements
- Operational precautions
- Waste treatment requirements
- Waste packaging and disposal requirements.

The characterization efforts will address both the 105-DR and 105-F Reactor buildings. Sampling and analytical activities have been separated into an initial scoping survey and four phases of sampling and analysis and radiological surveys. The project will generate four SAPs, one for each phase of the ISS project, following the initial scoping survey. The SAPs will include the specific human health and ecological risk based-values, based on Ecology (1996) formula values for soil, as appropriate for the constituents of concern. The human health and ecological risk-based values will serve as cleanup goals or verification criteria, depending on the phase. Each SAP will be sent to EPA and Ecology for review and approval prior to use in the field.

- Initial Scoping Survey: Initial characterization begins with a review of historical information, including the review of procedures, technical manuals, drawings, photographs, past radiological surveys, and interviews with experienced reactor operation personnel associated with the historical operation of the reactor. Environmental radiological scoping surveys and technical smears will be collected to identify the radiological conditions and isotopic distribution throughout the facility. A selected team of personnel will conduct facility inspections to examine the physical conditions of the facilities and determine suspect chemical/hazardous and radiological material locations. This information will be used to summarize the waste streams during project planning. The initial scoping survey does not require a SAP.

- Phase I: This phase included sampling and analysis efforts to support demolition activities scheduled for fiscal year 1998. The characterization activities specifically addressed the 105-DR lunchroom; miscellaneous storage room; switch gear room; and the 105-F lunchroom, shower room, storage/loadout area, and exhaust fan room. The characterization activities start with the development of a Phase I SAP (DOE-RL 1998d). After SAP approval, characterization continues with sample collection and laboratory analysis to support engineering design for removal and disposal of the components and structures to grade-level.

- Phase II: This phase includes all above-grade structures located outside the SSE at 105-DR and 105-F Reactor buildings. The sampling and analysis activities will support decontamination, removal, and disposal of all components and structures located outside the SSE to grade-level. Because characterization is used to provide insight on unknown, uncertain, and/or suspect conditions, characterization will be performed in conjunction with planned operations. Throughout the duration of this phase, facility conditions will change and additional information may become available that could alter the initial characterization plans. The Phase II SAP (DOE-RL 1998e) will address identified waste streams throughout the above-grade components and structures, as well as anomalies that may be found as the building layers are removed.

- Phase III: This phase, final status characterization, includes all grade-level concrete foundations, below-grade areas, and underlying soils excluding the 105-F FSB area. Specific objectives of this characterization activity include sampling and analysis of grade-level concrete foundations, below-grade concrete structures, and underlying soils to determine which materials are suitable for closure without decontamination. Characterization of the 105-DR fan room (part of the LSFF TSD unit) will be included in this phase. The characterization will also identify those areas that require removal and/or remediation prior to closure. This phase will include the development of a Phase III SAP (DOE-RL 1999), as well as a final verification report, which will document the status of the facility upon completion of the ISS projects. Although sampling specifications for the TSD unit may be included in a single SAP along with specifications for the CERCLA sites, RCRA closure requirements will be addressed.

- Phase IV: This phase, 105-F FSB material removal, includes sampling and analysis activities to support in-process characterization of material removal from the 105-F FSB. This phase also includes closure characterization activities in the 105-F FSB and surrounding area after the materials have been removed. Specific objectives of this characterization activity will include sample collection and laboratory analysis to support waste disposition of the soils and other material removed from the FSB. Characterization of the concrete structure and underlying soils will also be required to determine the extent of contamination and support the engineering design for decontamination, removal, and/or remediation of the concrete basin structure and soils. This phase will be implemented in accordance with an approved Phase IV SAP (DOE-RL 2000b~~a~~) and associated a~~A~~ddendum (DOE-RL-2000a~~b~~).

In addition to the four phases of characterization for the ISS projects, the 116-D, 116-DR, 117-DR, and 119-DR ancillary buildings/structures and associated ducting and piping within the 105-DR ancillary buildings will also be characterized. The characterization activities will follow the same general approach, to include the sample documents, as defined below. The characterization and verification sampling plan of these facilities will also require approval from Ecology but will be funded under separate cost accounts. The 116-DR and 117-DR, including associated piping, ducting, and tunnels, are currently part of a RCRA TSD unit and will be characterized following the appropriate RCRA protocols. Section 5.7 discusses closure of the TSD site in more detail.

4.3.1 Sampling Documents

4.3.1.1 Data Quality Objectives. The DQO procedure (BHI-EE-01, *Environmental Investigations Procedures,* Procedure 1.2, "Data Quality Objectives") will be used to define the quantity and quality of data to be collected to meet the project objectives. The results of the DQO process, including agreements, will be documented in a standardized DQO summary report, which will be developed prior to the SAP.

4.3.1.2 Sampling and Analysis Plan. Four separate SAPs will be prepared in accordance with BHI-EE-01, Procedure 1.15, "Sampling Documents." The SAPs will be written for each of the four phases identified in Section 4.3. The SAPs will be prepared in support of the Tri-Party Agreement (Ecology et al. 1998b) and site characterization, waste characterization, and/or environmental restoration and remediation activities. Each SAP will be comprised of a summary of the DQO, project-specific quality assurance project plan (QAPjP), and field sampling plan.

4.3.1.3 Quality Assurance Project Plan. When the DQO summary report has been completed and approved by the appropriate decision makers and technical team, the QAPjP will be written consistent with guidance provided in *Draft Final EPA Requirements for Quality Assurance Project Plans for Environmental Data Operations* (EPA 1997b). The QAPjP will address the following major elements:

- Project management
- Measurements and data acquisition
- Assessment and oversight
- Data validation and data usability.

4.3.1.4 Field Sampling Plan. The field sampling plan will be written following the completion of the QAPjP. This section of the SAP will describe the field sampling approach to be implemented in the field.

4.4 RESIDUAL RADIATION RELEASE CRITERIA

Underlying structures and soils will be disposed in accordance with Section 4.2 or will be released in accordance with criteria described in the action memorandum (Ecology et al. 1998a). The process to be used will be compliant with applicable Federal, state, and DOE-mandated requirements, such as those imposed by the CERCLA, the Tri-Party Agreement (Ecology et al. 1998b), and DOE Order 5400.5. The guidance document, *Guidance for Radiological Release of DOE Real Property at Hanford,* (DOE-RL 1997), will be used, where appropriate, to develop the appropriate scenarios for the derivation of radiological release limits based on D&D alternatives. The action memorandum (Ecology et al. 1998a) established a dose limit of 15 mrem/yr from all pathways. Additionally, the impact to groundwater may not exceed the MCLs.

The determination of whether to release the below-grade portions of the buildings will be based on the unrestricted residential use of the below-grade portions of the buildings. Dose to the receptors will be based on three risk scenarios: post drilling residential, sleeping resident, and building renovation (as outlined in DOE-RL [1997]). Radiological dose to the receptor is modeled via RESRAD-BUILD and RESRAD-SOIL computer codes supplied by Argonne

National Laboratory. RESRAD-SOIL is primarily used for estimating doses to a receptor from contaminated soil or other sources of contamination that may move into the soil and ultimately to groundwater. The contamination is modeled as a horizontal layer moving through soil. RESRAD-BUILD is used to derive the dose that an individual may receive while living, working, or visiting the released facility. RESRAD-BUILD may be used to create a three-dimensional model of a source term and facility (e.g., contaminated walls in a building). Both models consider multiple exposure pathways to the receptor (i.e., inhalation, ingestion, and external exposure).

RESRAD-SOIL has been widely used throughout the United States in modeling both radiological and chemical risk. RESRAD-BUILD has been used by several DOE sites in modeling radiological dose to workers and risk scenarios supporting release of real property. Both of the scenarios and software have been generally accepted by Washington State regulatory agencies (i.e., Ecology, DOT, and EPA). Parameters used in the model are outlined in DOE-RL (1997) and have been reviewed by these regulatory agencies. The scenarios were deemed acceptable for evaluation of release criteria for the 105-C ISS Project.

To proceed with close out, concentration-based limits must be developed that are based on the previously discussed scenarios and the 15 mrem/yr, 4 mrem/yr, and MCL limits. The target derived concentration guideline levels are based on land use, risk scenarios, and modeling. Verification measurements will be made after remediation, and the results will be used in the models to estimate the receptor dose from final radionuclide levels. Radiological measurement procedures are based on an approved SAP, and technical guidance provides regulatory requirements for the release of real and nonreal property. If the final runs of the dose model with the final measured radionuclide levels indicate that the dose limits are met, the data will be fully assessed and validated to ensure that the final verification data meets requirements.

All property that is released for offsite disposal and/or reuse and recycle is nonreal property. The release of nonreal property will follow the guidance in DOE Order 5400.5 and recent guidance (e.g., DOE 1995, DOE O 451.1A, DOE 1997, and BHI 1997c). For nonreal property that is surface-contaminated only (not contaminated in volume), the project may use the processes described in the above referenced documents and authorized limits that are provided in Table 1 of DOE (1995). If the property meets the surface contamination limits and the person or entity receiving the property is aware of the measured radioactivity on the property, the property may be dispositioned without regard to residual radioactivity.

If the property is volumetrically contaminated (e.g., soil, concrete exposed to water, activated concrete, and activated metal), the property must be evaluated through a pathway dose assessment to determine the potential dose to a receptor and residual contamination allowable. Depending on the estimated dose to a member of the public, DOE guidance (DOE 1995) requires U.S. Department of Energy, Richland Operations Office (RL) and/or DOE-Headquarters approval, as well as the concurrence of the property recipient (e.g., the state or disposal site operator) for any nonreal property disposed as waste. An ALARA evaluation must be performed prior to final use of any surface or volumetric authorized limits. The DOE has provided guidance for ALARA evaluations applied to environmental releases in a recent document (DOE 1997).

4.5 NATURAL AND CULTURAL RESOURCES PROTECTION

Most of the areas surrounding the 105-F and 105-DR Reactor buildings are covered with graveled and asphalt surfaces. Native vegetation does not exist within the reactor exclusion areas; therefore, no avoidance mitigation will be required to protect native plants. An ecological and cultural review of the 105-DR and 105-F Reactor buildings and surrounding areas was conducted ~~on February 26~~ in March and April~~-15~~, 1998~~;~~ to identify cultural and ecological resources (Teel 1998a; Brandt 1998a~~;~~, ~~Brandt~~ 1998b).

The findings of the ecological review of the 105-F Reactor building indicated that bats use the building during the summer months, as evidenced by the presence of bat feces and several dead Pallid bats (i.e., a Washington State monitor-2 species). No accumulations of bat feces or any other evidence were observed that indicated use of the reactor by roosting concentrations of bats. However, artificial bat roosts were installed around the perimeter of the reactor as mitigation for potential loss of roosting habitat. The review concluded that no adverse impacts would occur to the bats if demolition activities began outside their active season (which is approximately April through October) and continued to be uninterrupted. If demolition is scheduled to begin during the bats' active season or is interrupted for a period of time sufficient that bats could commence using the building for roosting (i.e., from several weeks to several months), a new ecological review would be required.

~~An ecological review of the 105-DR Reactor building was conducted over four separate visits, between February and April 1998 (Brandt 1998a).~~ The findings of the~~is~~ ecological review for the 105-DR Reactor building revealed the presence of bats using the facility for roosting. Two live specimens of small-footed Myotis bats were found within the reactor building, and numerous small deposits of scattered feces were found throughout. The small-footed Myotis is a former Federal candidate species for threatened and endangered status and is currently listed by the State of Washington as a priority species where it occurs in natural breeding areas and other communal roosts. No evidence of a communal roost or large aggregation of bats was found within the reactor building. However, a communal roost was found within the process water tunnel that enters the reactor from the 190-D (now demolished) water plant. This communal roost had been previously documented (Becker 1993) and appeared to still be active.

The review concluded that no adverse impacts would occur to bats using the reactor building if demolition activities began outside their active season (which is approximately April through October) and continued to be uninterrupted. If demolition is scheduled to begin during the bats' active season or is interrupted for a period of time sufficient that bats could commence using the building for roosting (i.e., from several weeks to several months), a new ecological review would be required.

The review also concluded that the bats using the 190-D and 190-DR process water tunnels for roost sites would be impacted and the habitat would be lost because the project would isolate the tunnels from the reactor, thereby closing off access to the tunnels. Therefore, a mitigation plan was proposed and implemented that -preserves the tunnels for habitat by creating an alternate opening at one of the existing surface hatches of each tunnel and installing "bat gates" that would allow access to the bats and exclude access to people and other animals (Teel 1998b). Other mitigation strategies will be evaluated and implemented, as appropriate, throughout the life of the project.

No other species of concern were identified either inside or in areas surrounding the 105-DR and 105-F Reactor buildings that would be impacted by the projects ~~at 105-F and 105-DR Reactor buildings~~. However, because of the possibility of migratory birds using the buildings and surrounding areas for nesting, annual surveys (and seasonal surveys, as needed) would be required throughout the duration of this project. If nesting migratory birds are found within the project area, mitigation actions will be developed to avoid nest destruction or abandonment.

The areas surrounding the 105-F and 105-DR Reactor buildings will be maintained clear of vegetation until all associated waste sites have been remediated and final disposition of the reactors occurs. Therefore, revegetation of these sites is not planned to immediately follow the ISS projects. However, provisions will be made during final site grading to ensure that suitable soils are in place to facilitate revegetation at a later time.

~~A follow-up ecological review of the 105-DR and 105-F Reactors was conducted on April 27, 2000[2]. Because of the mitigation efforts already taken at these areas for bats, this survey was conducted to identify any potential bird species of concern nesting on the outside of the structures and surrounding areas. At the 105-F Area, the following birds were observed: two pairs of rock doves (*Columba livia*), three pairs of European starling (*Sturnus vulgaris*) on the reactor building, and one pair of house finches (*Carpodacus mexicanus*) in a sycamore tree near the office trailers. The area surrounding the reactor building is bare ground with no vegetation. No species of concern were identified within the active project area, and no impacts to ecological resources are anticipated from continued activities at 105-F Reactor.~~

~~At the 105-DR Reactor, the following birds were observed: six pairs of starlings on the reactor building, two pairs of house finches, one Say's phoebe (*Sayornis saya*) on the fence near the lay-down area, one cliff swallow (*Petrochelidon pyrrhonota*) flying high above the reactor, and two white-crown sparrows (*Zonotrichia leucophrys*) on the ground near the office trailers. No species of concern were observed within the active project area and no impacts to ecological resources are anticipated from continued activities at 105-DR Reactor.~~

~~House finches and Say's phoebes have been known to nest in areas similar to the lay-down area. These species as well as nearly all other species except starlings, rock doves, house sparrows, and upland game birds are protected by the Migratory Bird Treaty Act during nesting and~~

[2] ~~Email, KA Gano to SM Hamblin, *Ecological Review Update for 105-DR & 105-F Reactors (00-ER-010)*, CCN #078686, dated April 27, 2000.~~

~~destruction of their nest and occupants is not allowed. Therefore, if any nesting birds other than starlings or pigeons are threatened by project activities, BHI Natural Resources and Risk Assessment will be contacted to provide instructions on mitigation.~~

Cultural resource reviews ~~(BHI 1998e, 1998f) were conducted on January 14, 1998, at the 105-F Facility and February 7, 1998, at the 105-DR Reactor building that~~ at the 105-DR and 105-F Reactor buildings considered cultural and historic significance. ~~Also, a cultural resources field reconnaissance was conducted on February 26, 1998.~~ These reviews concluded that the 105-DR and 105-F Reactors are situated in areas of low cultural resource potential and that no archeological resources have been recorded within or adjacent to the areas of potential effect. Due to the extensive disturbance to the areas, no intact subsurface materials are anticipated. However, due to potential interest from Native American Tribal representatives to monitor ground-disturbing activities associated with demolition, notification will be provided one week prior to ground-disturbing activities.

The reviews of historic significance found that both the 105-DR and 105-F Reactors were determined by RL and the Washington State Historic Preservation Office to be contributing properties within the Hanford Site Manhattan Project and Cold War Era Historic District. However, neither reactor was selected for individual documentation, and additional mitigation measures were not required prior to demolition. Pursuant to Stipulation V (C) of the Historic Buildings Programmatic Agreement (DOE-RL 1996), a walkthrough of the 105-DR and 105-F Reactor buildings was conducted wherein ~~on April 29, 1998, to determine which, if any, artifacts and objects remaining within these buildings had educational or interpretive potential. Fifteen~~ several items were identified, ~~and~~ tagged, and ~~which will be~~ relocated to a secure area for storage prior to the initiation of project activity.

These initial ecological and cultural reviews have been periodically updated (Teel 2001a~~;~~, ~~Teel~~ 2001b), and specific information can be found in these referenced reports. ~~Natural~~ Ecological and cultural resource reviews for the ancillary facilities ~~have not been conducted and will need to be~~ are completed prior to initiating D&D activities at these locations. Ecological and cultural reviews for ongoing activities will be conducted at least annually.

4.6 ADDITIONAL RELEVANT CONTROLS OR ACTIONS

The following special controls are defined in the 105-DR Auditable Safety Analysis (ASA) (BHI 1998b) and applicable MOC documents, and will apply to activities conducted at 105-DR:

- To ensure that the conditions assumed in the hazard analysis are maintained, no intrusive activities will be performed on the reactor block.

The following special controls are defined in the 105-F Auditable Safety Analysis (ASA) (BHI 1998c) and applicable MOC documents, and will apply to activities conducted at 105-F[3]:

- No activities will be conducted that require penetrating the gas cover shell of the reactor block. Water and gas piping may be cut and capped if a JHA is performed, a hot work permit is in place, and radiological control requirements are met. The potential flammability of the masonite in the biological shield will be considered while preparing the required permitting, and pathways for heat and burning to the material will be addressed.

- No bottles of pressurized flammable material will be stored in the basin until after all potential fuel elements have been excavated and secured within appropriate shipping containers. The wooden planking and other flammable materials that are uncovered during remediation of the FSB will be removed and staged away from the basin.

- Excavation of the soil from the FSB will be carried out until approximately the last 1 m (3 ft) of soil remains in the FSB to provide basin workers with the shielding properties of the soil (which is how high the top of the concrete spacers on the floor of the FSB reach). The last 1 m (3 ft) is where items would have been collected. Surveys of the surface of the soil will be made to identify areas with elevated dose readings. These locations will be identified as possible areas where high-dose-rate items (including pieces or whole fuel elements) may be located.

- Following identification of areas with elevated surface dose rates, material may be stored to keep radiological doses ALARA until packaging is performed. Remote-operated devices and/or long-handled tools will be used to remove the covering soil and debris over the high-dose-rate items, extract the items from the excavation site, and deposit these items into the appropriate shipping containers (cutting where appropriate and/or required). This operation will be performed for each location containing potential high-dose-rate items.

- Excavation and removal of high-dose-rate items from one location in the basin will be completed before another site is to be excavated. This is to control direct doses from multiple sources. Appropriate measures to control doses to workers will be used, typically by controlling the duration of worker exposure to external doses, using various types of shielding, and controlling the distance of the operator to the high-dose-rate items.

[3] During the same time frame of the revision of the RAW document from Revision 0 to Revision 1, a special control was being deleted from the ASA document. The MOC document (MOC-2001-0002) addresses this matter and is anticipated to be approved at approximately the same time as the revision of the RAW. The special control reads, "The concrete spacers on the FSB floor will not be removed until all potential high-dose-rate items have been excavated and secured with appropriate shipping containers."

5.0 PROJECT MANAGEMENT AND ORGANIZATION

5.1 PROJECT SCHEDULE AND COST ESTIMATE

The 105-DR and 105-F ISS Projects have been scheduled and estimated using the ERC hierarchy of schedules, which include activity logic and restraints. Activities will be resource loaded for both nonmanual and manual personnel. Equipment needs are identified and other materials are estimated and included in the budgeted cost of work scheduled.

Estimates of project costs have been prepared at the activity level by the project team and subsequently have been reviewed and approved by the ERC, RL, EPA, and Ecology.

The removal action was initiated in August 1998. The current schedule, which encompasses the work scope of the 105-F and 105-DR ISS Project and ancillary buildings ~~(beginning in fiscal year 1999 through project completion),~~ is included in Appendix A. Appendix A also includes the project scope for all of the projects covered under this removal action by fiscal year~~, with associated dollars and general assumptions~~. A more detailed schedule, including assumptions, resources, and activity breakdown, will be developed and submitted with the detailed work plan each fiscal year. The major activities for which a roll-up of cost and schedule performance has been prepared include the following:

- Auditable safety analysis/final hazard classification document preparation
- DQO/SAP
- EE/CA and action memorandum
- Air monitoring plan and best available radionuclide control technology documentation
- Pre-construction, mobilization, and support
- Radiation scope identification surveys
- Decontamination
- Asbestos abatement
- Hazardous material removal
- Pipe and equipment removal
- Structure demolition
- Radiation monitoring/sampling/analysis
- Equipment/material/consumables
- SSE subcontract/construction
- Demobilization/project closeout.

Schedule status is reviewed weekly in 105-DR and 105-F ISS Project review meetings. On a monthly basis, cost and schedule performance are reviewed by the ERC. Members of DOE, EPA, and Ecology are invited to participate in these review meetings.

5.1.1 Project Cost and Schedule Tracking

Performance measurement and analysis is performed by the D&D Project Planning and Controls organization. Project cost and schedule are controlled and updated using the ERC Management Control System, as described in BHI-PC-01, *Baseline and Funds Management System.*

An earned-value system tracks the cost, schedule, and performance for all D&D projects as they progress towards completion. Cost/schedule performance reports provide budgeted cost of work scheduled comparisons and budgeted costs of work performed against the actual cost of work performed. These reports provide variances to the baseline schedule and cost as budgeted in the project's detailed work plan. Variances above threshold values are documented, as well as the rationale for the variance(s) and any recovery plan required.

Trends and baseline change proposals are readily identified through the ERC formal trend and change control program (BHI-PC-01, PCP 1.11, "Trend Identification, Monitoring, and Analysis," and PCP 1.12, "Baseline Change Control"). All changes that affect the baseline are documented. The ERC trend register, which is reviewed monthly by ERC senior management, categorizes trends from conception to final resolution. Trends are identified as either performance trends or scope trends and are further defined as resolved or unresolved.

Fiscal year project staffing, as budgeted, is reconciled monthly during the project reviews to the actual number of full-time-equivalent personnel used during the month. Likewise, the corresponding number of hours actually worked are presented and compared to the budgeted current work plan. Actual overtime is monitored monthly (by department) and reconciled to the current budgeted overtime.

Cost and schedule variances to the current budget are tracked both on a monthly and to-date basis and are reconciled back to the cause of the variance. Project impacts due to the cost and/or schedule variance are described and corrective actions are identified and tracked to the point of final resolution.

5.2 CONDUCT OF OPERATIONS

Conduct of operations is imposed to ensure that work is performed in a controlled and organized manner, that all facets of work activities have been considered, and that necessary documentation is maintained.

The performance of field activities and D&D projects is governed by *Decontamination and Decommissioning Project Manager's Implementing Instructions* (PMII) (BHI 1998a), applicable field support instructions, and specific work instructions. The PMII is based on a graded approach to the conduct of operations authorized by DOE Order 5480.19 and the ERC D&D conduct of operations applicability matrix. The PMII is applicable to all ERC personnel, assigned or matrixed, who perform activities under the responsibility and direction of the D&D project manager. The applicability matrix is issued and maintained by the D&D project manager and identifies elements of the DOE order that apply to project activities, the implementing documents, and any deviations or exceptions to the DOE order guidelines.

Conduct of operations strongly emphasizes technical competency, workplace discipline, and personal accountability to ensure the achievement of a high level of performance during all activities. Project personnel are responsible for fully complying with the PMII; if conflict arises with other instructions or directions, work will be safely stopped until resolution is achieved. Safety is the first priority, and all planning will include appropriate safety analyses to identify potential safety and health risks and the means to appropriately mitigate them. Workers will not start work until approved safety procedures, instructions and directions, which implement the Integrated Environment, Safety, and Health Management System, are provided.

Conduct of operations requires workers to be alert and aware of conditions affecting the job site. Operators and workers conducting field activities should be notified of changes in the building and/or work area status, abnormalities, and difficulties encountered in performing project operations. Similarly, operators and workers will notify the chain of command of any unexpected situations. In accordance with the severity of a finding (i.e., emergency condition), notification requirements will be expanded to include upper-tier management and regulatory agencies.

5.3 CHANGE MANAGEMENT/CONFIGURATION CONTROL

If a change arises that results in a fundamental change to the selected response action that is not within the scope of the action memorandum (Ecology et al. 1998a) and the implementing documents, then an EE/CA or proposed plan and supporting documentation will be prepared to allow DOE, EPA, and Ecology to select a revised response action.

Established configuration/change control processes ensure that proposed changes are reviewed in relation to the specified commitments. In the event that discovery indicates a breach of these commitments, work ceases so stabilization and/or recovery actions may be identified and implemented as appropriate. The BHI off-normal event procedures describe the reporting process and protocol applicable to such a discovery. BHI-DE-01, EDPI 4.40-01 defines the MOC process for facilities that have a final hazard classification of less than nuclear. The management of change process is used as follows:

- Evaluate the impact of proposed changes that could affect authorization basis documents.
- Determine whether proposed changes require prior DOE approval.
- Evaluate the impact of discovered conditions.
- Evaluate the effect of deviations from activities or commitments described in authorization basis documents.

5.4 PERSONNEL TRAINING AND QUALIFICATIONS

During the performance of project activities, the experience and capabilities of the operating staff are extremely important in maintaining worker and environmental safety. Day-to-day knowledge of ongoing operations, month-to-month understanding of conditions encountered, and lessons learned will be vital to continued safe operations.

Training requirements will ensure that personnel have been instructed in the technologies to work safely in and around radiological areas, and to maintain their individual radiation exposure and the radiation exposures of others ALARA. Standardized core courses and training material will be presented, and site-specific information and technologies will be added to adequately train workers.

Health physics workers are required to have completed and be current in radiological control technician qualification training. These training courses require the successful completion of examinations to demonstrate understanding of theoretical and classroom material.

Specialized training will be provided as needed to instruct workers in the use of nonstandard equipment, in the performance of abnormal operations, and in the hazards of specific activities. Specialized training may be provided by on-the-job training activities, classroom instruction and testing, or pre-job briefings. The depth of training in any discipline will be commensurate with the degree of the hazard(s) involved and the knowledge required for task performance.

Some activities will require the acquisition of expert services as opposed to project staff training. Assaying of waste packages and dismantling the facility by specialized methods (e.g., diamond wire sawing) are examples of activities requiring expert assistance.

The ERC Training Program provides workers with the knowledge and skills necessary to safely execute assigned duties. A graded approach is used to ensure that workers receive a level of training commensurate with their responsibility that complies with applicable requirements. Specialized employee training includes pre-job safety briefings, plan-of-the-day meetings, and facility/work site orientations. The following training and qualifications may be applicable as required by job assignment for work activities:

- Training in accordance with 29 CFR 1910.120
  - 40-Hour Hazardous Waste Worker/8-Hour Refresher
  - 24-Hour Experience Component
  - 8-Hour Supervisor Training (for selected individuals)
  - SS HASP and RWP
  - Respirator Training
  - First Aid (two qualified persons per shift/crew)
  - Certified Asbestos Worker and/or Asbestos Awareness
  - Lead Worker
  - Radiation Worker II.

- Medical surveillance requirements
  - Hazardous waste worker physical
  - Mask fit
  - Lead worker baseline
  - Asbestos worker.
- Dosimetry and bioassay requirements
  - Thermoluminescent dosimeter (as directed in the RWP)
  - Plutonium bio-assay (as determined by the Radiological Control organization)
  - Whole body count.

The SS HASP, RWP, and JHA will include specific requirements for project activities being conducted, which include PPE and required training for project personnel. This is discussed in detail in Section 3.2.

5.5 PLAN FOR READINESS

A readiness assessment will be conducted, in accordance with BHI-MA-02, *ERC Project Procedures*, Section 8.2, "Readiness Assessments," to detail the level of readiness that will be required to initiate certain project activities. The project manager is responsible for defining activities requiring a readiness assessment and the type required in accordance with the readiness assessment procedure. The decision to conduct a readiness assessment for certain work scopes will be based on the risk involved and the complexity of the work or work outside the boundaries of the authorization basis. A readiness assessment will also be required if the project activities are shut down for reasons other than routine, as described in the readiness assessment procedure.

5.6 QUALITY ASSURANCE REQUIREMENTS

The overall quality assurance for the removal action work plan will be planned and implemented in accordance with DOE O 414.1A, 10 CFR 830.120, and other applicable standards. Full implementation will be required when the conceptual design for the SSE is approved and detailed design is initiated. The quality assurance activities will be graded based on the potential impact on the environment, safety, health, reliability, and continuity of operations. Specific activities include quality assurance implementation, responsibilities and authority, document control, quality assurance records, and audits. These activities are discussed in the following subsections:

5.6.1 Quality Assurance Implementation

All project-related activities will establish and implement appropriate quality assurance requirements. Conditions adverse to quality will be identified in nonconformance reports, audit reports, surveillance reports, and corrective action requests. Investigation and corrective actions in response to these adverse conditions will be completed in a timely manner.

5.6.2 Responsibilities and Authority

BHI must perform quality engineering, design reviews, surveillance, and audits (as necessary) to achieve quality assurance objectives. BHI must also ensure that the various contractors and design agencies establish design and D&D quality assurance programs to control design and D&D in accordance with applicable requirements. The D&D contractor(s) must establish, implement, and document an inspection plan in accordance with approved specifications and drawings.

5.6.3 Document Control

The D&D documents, such as specifications and drawings, will be controlled in accordance with approved configuration management procedures. The responsible design agency will maintain control of the design D&D documents through acceptance of the documents. A project records checklist will be initiated to identify those records required for the final project file.

5.6.4 Quality Assurance Records

Each organization that maintains quality assurance records will be required to control the records in accordance with applicable BHI quality assurance requirements.

5.6.5 Audits/Assessments

Internal and external audits are to be performed by the Compliance and Quality Programs organization to ensure project compliance with the quality assurance program requirements.

5.6.6 Self-Assessments

Self-assessments will be conducted by project personnel to determine compliance in accordance with the requirements of BHI-MA-02, Procedure 2.7, "Self-Assessment."

5.7 PROJECT CLOSEOUT

Removal of the facilities (up to reactor shield walls) and their systems will be completed to a minimum depth of 0.9 m (3 ft) below grade. At this excavation level, additional characterization will be conducted to verify the status of the remaining below-grade structure. If the remaining below-grade structure is found to exceed cleanup standards, then excavation will continue to a maximum depth of 4.6 m (15 ft) below grade. If groundwater protection standards are not met at the 4.6 m (15 ft) depth, then any additional remediation will be coordinated with and conducted under the remedial action for the 100-DR-2, 100-DR-1, and 100-FR-1 Operable Units.

Soil excavated as a result of the removal of below-grade structures will be designated and disposed, as appropriate.

Waste sites within the footprint of the facilities will be remediated. If large volumes of contaminated soil are encountered or if removal of the contaminated material inhibits reactor safe storage activities and/or are not cost effective, then remediation of these sites will be coordinated with and conducted under the final remedial action for the 100-DR-2, 100-DR-1, and 100-FR-1 Operable Units. The lead regulatory agency will be involved in determining the cost effectiveness and deferrals to the RA/WD Program. The impacted waste sites will be covered with a cap of clean borrow soil and routine S&M activities will be initiated. The S&M efforts and remediation planning will be consistent with that established for the 100 Areas until remedial action in the 100-DR-2, 100-DR-1, and 100-FR-1 Operable Units. Waste sites that are remediated and achieve the established cleanup standards during the removal action will be documented and removed from the Waste Information Data System in accordance with Tri-Party Agreement Procedure TPA-MP-14 (Ecology et al. 1998b).

Upon completion of D&D activities, a minimum of 0.9 m (3 ft) of clean fill/soil cover will be placed over any remaining below-grade structure and inert/demolition material, and the fill will be graded to match the surrounding terrain.

The ARARs for project activities identified in the action memorandum (Ecology et al. 1998a) establish the cleanup criteria for the 105-DR and 105-F sites. Cleanup levels specified for soils, below-grade structures, and fill materials will meet the MTCA Method B standard for nonradiological contaminants in soils and for radionuclides the EPA protectiveness factor of 15 mrem/yr effective dose equivalent to 4.6 m (15 ft) below grade. These cleanup levels will also ensure that groundwater MCLs will not be exceeded. Rubble created from demolition of the structure will be evaluated against dangerous waste criteria, ERDF waste acceptance criteria (BHI 1998d), and radiological standards. Materials that exceed dangerous waste criteria radiological standards will be appropriately segregated and disposed in accordance with Section 4.2. Subsurface structures and debris that meet the MTCA and radiological standards may be left in place.

Closure requirements for the 105-DR LSFF RCRA TSD unit will be met by implementing this removal action. Clean closure of the TSD unit will be verified through sampling and will be documented in a cleanup verification report.

After completion of all demolition activities, a project closeout report will be prepared. The report will include the location and quantities of waste dispositioned, project costs, lessons learned, summarization of characterization and monitoring data, and a reconciliation to the conceptual model for D&D baseline estimating. The report will be forwarded to the records retention center where it will be included in the Administrative Record for the 100-DR-2, 100-DR-1, and 100-FR-1 Operable Units. The report will also be included in the TSD unit-specific operating record for the 105-DR LSFF through transmission to BHI Document and Information Services in accordance with procedures established in Section 6.2 of the *RCRA Permit Implementation Plan* (BHI 1997a). In accordance with the action memorandum (Ecology et al. 1998a), the closeout report will be submitted to the regulatory agencies for review and approval.

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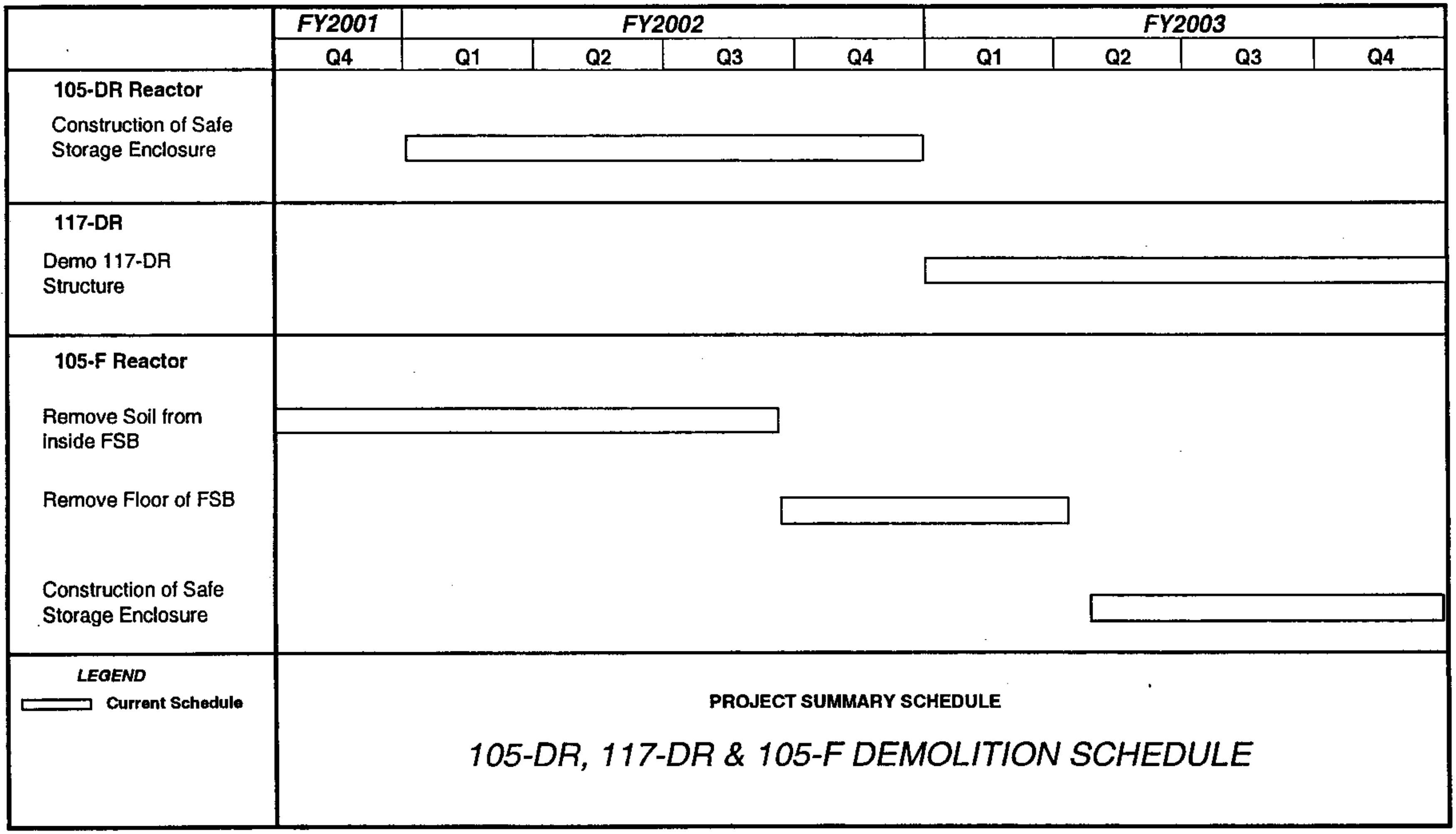
APPENDIX A

105-DR AND 105-F INTERIM SAFE STORAGE PROJECT SCHEDULE

Appendix A – 105-DR and 105-F Interim Safe Storage Project Schedule



APPENDIX B

AIR MONITORING PLAN

APPENDIX B

AIR MONITORING PLAN

B.1 INTRODUCTION

The decontamination and decommissioning (D&D) activities for the 105-DR and 105-F Reactor buildings, 116-D and 116-DR Reactor exhaust stacks, 119-DR Exhaust Air Sampling Building, 117-DR Exhaust Filter Building, and associated above-grade ducting are initial steps in placing the 100 Area reactor buildings into interim safe storage (ISS). All of these D&D activities have been identified as *Comprehensive Environmental Response, Compensation, and Liability Act of 1980* (CERCLA) program activities (DOE-RL 1998). Quantification of radioactive emissions, implementing best available radionuclide control technology (BARCT), and air monitoring have been identified as substantive requirements (i.e., relevant and appropriate requirements). A BARCT compliance demonstration is determined by the regulatory agency on a case-by-case basis. These substantive requirements are in accordance with *Washington Administrative Code* (WAC) 246-247-040. This appendix presents compliance with those requirements.

A key assumption of previous revisions to this plan was that no more than five spent fuel elements would be found during cleanout of the 105-F Fuel Storage Basin (FSB) (BHI 1998c). This was based on previous experience with the deactivation and cleanout of the FSBs at 105-D and 105-DR. A total of five spent fuel elements were found between these two basins (three at the 105-DR FSB and two at the 105-D FSB). As the project has progressed with cleanout activities at the 105-F FSB, additional elements have been located. This prompted a need to revise the inventory assumptions, the associated potential to emit, and the unabated off-site dose calculations. These changes were originally documented and approved in an addendum to this plan (Henckel 2002) and are being included as part of this revision.

B.1.1 Planned Activities

B.1.1.1 105-DR and 105-F Reactor Buildings. The D&D activities on the 105-DR and 105-F Reactor buildings have the potential to generate particulate radioactive air emissions. The construction of the safe storage enclosures (SSEs) will not have a potential-to-emit (PTE) radioactive air emissions because most of the contamination (to a reasonable extent) will be fixed or removed during D&D. Both reactor building D&D activities will be performed during the same time period (in parallel). For each facility, the four areas of concern for air emission releases are the following:

- Radiological buffer areas
- Exhaust plenum
- Valve pit, process area, and solids feeds area
- Inner and outer rod rooms (see Section 1.2 of this document for a description of these areas).

The removal of contamination ranges from nonaggressive to aggressive decontamination. Nonaggressive decontamination includes methods such as wiping or applying fixatives to stabilize contamination. Aggressive decontamination techniques are methods such as scabbling, abrasive blasting, vacuuming, and the use of portable ventilation filter units. If these techniques do not remove all the contamination or it is not cost effective to do so, the contamination will be fixed with a fixative or the contaminated material will be removed and disposed as appropriate.

Demolition methods will be selected based on the structural elements to be demolished, remaining radionuclide contamination, location, and integrity of the SSE. Such methods could include use of an excavator with a hoe-ram, a hydraulic shear with steel shear jaws, concrete pulverizer jaws or breaker jaws, or a crane with wrecking ball. Any fixed contamination will be segmented using techniques such as cutters or mechanical/power saws and will be handled separately.

B.1.1.2 116-D, 116-DR, and 119-DR and Ducting. The D&D activities performed on the 116-D and 116-DR stacks, 119-DR Building, and associated above-grade ducting have the potential to generate particulate radioactive air emissions. The D&D activities are anticipated to be performed on the 116-DR and 116-D exhaust stacks, 119-DR Exhaust Air Sampling Building, and associated above-grade ducting in the last quarter of fiscal year (FY) 1999 and the first quarter of FY 2000. Nonaggressive decontamination includes methods such as wiping or applying fixatives to stabilize contamination. Aggressive decontamination techniques include methods such as scabbling, abrasive blasting, vacuuming, and the use of portable ventilation filter units. If these techniques do not remove all of the contamination or it is not cost effective to do so, a fixative will be applied or the contaminated material will be removed and disposed as appropriate.

The D&D of the 116-D and 116-DR *stacks will include stabilization of contamination in the* lower interior portions of the stack, vacuuming of loose debris inside the stack, drilling into the stacks to place explosives, removing ducts, falling the stack with explosives, rubblizing the upper 0.91 m (3 ft) of the foundation with explosives, and reducing the size of the concrete rubble for disposal at the Environmental Restoration Disposal Facility (ERDF). The areas that will be drilled for placement of explosives will be decontaminated before placing the explosives. Stabilization of the lower portions of the stack may consist of wiping, vacuuming, and/or applying fixatives. Before demolition, if loose debris on the interior floor of the stack is present, a high-efficiency particulate air (HEPA) vacuum may be used to vacuum the debris. After the stack is demolished, waste verification sampling will occur, followed by size reduction with heavy equipment, to prepare for disposal at the ERDF.

Standard demolition techniques (i.e., excavator with bucket or hydraulic shears) will be used to demolish the 119-DR Building and associated above-grade ducting. If decontamination is necessary, methods such as wiping, vacuuming, and/or applying a fixative will be used.

B.1.1.3 105-F Fuel Storage Basin. A fuel storage basin (FSB) disposition plan (BHI 1999a) was developed after various alternatives and removal methods were reviewed. The disposition plan discusses the cleanout of the FSB (as part of facility ISS) and how cleanout will be completed in two stages. BHI (1999a) emphasizes Stage I, with general approaches for Stage II removal activities. The method chosen for basin cleanout combines safety and cost effectiveness using standard techniques, available equipment, and proven technologies. Initially, a water-removal system will be fabricated, installed, and operated within the FSB. Water will be removed, tested, and disposed of at the 200 Area Effluent Treatment Facility. Stage I will address the FSB from the surface to approximately 5.2 m (17 ft) below grade. Stage II will address the remaining approximately 0.9 m (3 ft), where the radioactive sediment and potential fuel elements exist. Both phases will apply engineering controls to ensure the safety of workers, maintaining radiological doses as low as reasonably achievable (ALARA), and minimizing potential releases of radiological emissions to the environment. A readiness assessment will be conducted to assess project readiness prior to initiating FSB cleanout work. A major portion of material/structure removal will use equipment (e.g., backhoes and excavators) that is typically used for D&D below-grade material removal, structure demolition, and large-scale soil remediation work. Experience gained in past basin cleanout projects (i.e., 105-B, 105-C, 105-D, 105-DR, and 105-N) will be used to optimize the working conditions.

B.1.1.3.1 Upper Fill Removal (Stage I). Demolition of the above-grade structure will occur to allow access to the upper fill overburden and characterization of the fill material. Excavation of backfill will begin from grade and continue to approximately 5.2 m (17 ft) below grade using primarily an excavator. The FSB columns will also be demolished during the excavation of the top 5.2 m (17 ft) of backfill. The frequency of radiological surveys will increase as necessary as removal of backfill material approaches the bottom of the basin. After being sampled and deemed appropriate for reuse, the upper fill material will be used as backfill upon U.S. Environmental Protection Agency approval. The remaining soil will act as a biological shield and will continue to confine the sediment and other radiological material located on or near the basin floor. Originally, a retractable cover over the FSB was planned to be installed and used during extended periods of work inactivity or severe inclement weather conditions. The need for such an enclosure was later reevaluated. Phase II excavation and demolition activities (Section 2.1.4.2) would be significantly easier and more efficient to perform without a cover and supporting structures. Unobstructed access to the basin allows for easier deployment of equipment. Supplemental lighting is minimized or eliminated. Subsequently, DOE and the lead regulatory agencies agreed that a cover would not be necessary (Smith 2000).

B.1.1.3.2 Debris and Contaminated Fill Removal with the Integrated Equipment System (Stage II). Removal of fuel elements or hot spot materials will typically be accomplished using a small excavator (e.g., Brokk™-type remote control crawler) with attachments to expose the fuel elements or hot spot materials and then load materials into an appropriate container. A large excavator or personnel with long-handled tools may be used to remove hot spot material and other lower fill debris and soil, if dose rates do not pose a threat to personnel. Loaded fuel containers will be placed into appropriate shipping containers prior to removal from the controlled area. Other hot spot materials will be excavated, and packaged for disposal. Backfill material removed during Stage II will be radiologically screened and any fuel elements will be

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remotely placed into a fuel holding container. Other hot spot materials will be removed and packaged for disposal.

Heavy equipment will provide shielding and distance for operators, allowing for prudent excavation of high-dose-rate material. The soil/debris material (except for retrievable spent nuclear fuel) will be packaged for disposal at ERDF or the Central Waste Complex, as appropriate. Some waste materials may require segregation for macro-encapsulation, such as lead and/or high-dose materials (e.g., sediment) to be accepted for disposal at the ERDF (BHI 1998a). Typically, loader track hoe, backhoe bucket, and attachments will hook, grip, shear, lift, or remove buckets, monorail beams, wood planking material, etc., and place the materials into ERDF-approved containers.

Demolition of the walls, floor, and underlying soils would occur during Stage II, depending on characterization results. RESidual RADioactivity dose model methodology may be used to determine radiological release levels for the basin walls, concrete floor, remaining structures, and adjacent (beside and below) soils, similar to that used for the 105-C FSB and below-grade structures.

B.1.1.4 117-DR and Associated Tunnels. The 117-DR Exhaust Filter Building and associated underground tunnels (between the 105-DR and 117-DR and between the 117-DR and 116-DR) are not included in this monitoring plan. When additional information becomes available, this plan (including air monitoring) will be updated, as appropriate, and submitted to the lead regulatory agencies for approval.

B.2 AIRBORNE SOURCE INFORMATION

B.2.1 105-F and 105-DR Reactor Buildings

The potential exists for particulate radioactive airborne emissions resulting from the D&D activities previously described. Emission estimates were based on the radiological surveys and 105-C characterization information. The surface area and the contamination levels for each area of the facilities were based on the radiological characterization surveys performed to date at the 105-DR and 105-F Reactor buildings (radiological survey records [RSRs] 105F-98-0001 through 105F-98-0182, and 105DR-98-0007 through 105DR-98-0051). Because these radiological surveys did not provide a distribution of radionuclides, the radionuclide distribution was derived based on the 105-C Notice of Construction (DOE-RL 1996), with the addition of a few isotopes. The primary radiological composition consists of americium, cobalt, cesium, europium, strontium, and plutonium. As stated in the 105-C Notice of Construction (DOE-RL 1996), the radionuclides that were considered to be insignificant dose contributors (i.e., less than 1% of the total) were not listed. The radiological concentrations for americium, cobalt, cesium, europium, and strontium were taken from 105-C FSB sample data, which is considered one of the most volumetrically contaminated area within the 105-C Reactor building. The plutonium values were estimated based on information found in *The Technical Basis for Internal Dosimetry at*

Hanford (PNL 1991). The isotopes selected bound the possibility of what can be found, therefore creating a conservative estimate.

The radionuclide annual possession quantities and subsequent potential emission calculations are presented in Table B-1 for the D&D of the 105-DR and 105-F Reactor buildings. It is assumed that the airborne contamination released (i.e., the PTE value) from each facility will be the same. However, the unabated dose rates will be different because the facilities are located at different areas on the Hanford Site. Additionally, it was assumed that 95% of the total annual possession quantity will represent the emissions from nonaggressive methods, and the release fraction of 1E-03 will be applied in accordance with *Washington Administrative Code* (WAC) 246-247-030(21)(a). Five percent of the total annual possession quantity will represent the emissions from aggressive methods, and the release fraction of 1 will be applied per direction provided by the Washington State Department of Health (Conklin 1994). Details of the calculations are documented in Bechtel Hanford, Inc. (BHI) calculation 0100X-CA-V0020, *105 DR & F Demolition Activities* (BHI 1998b).

Table B-1. 105-DR and 105-F PTE Values.

Radionuclide	Annual Possession Quantity (Total) [a] (Ci/yr)	PTE (1E-3 RF)[b] (Ci/yr)	PTE (1RF) [c] (Ci/yr)	PTE (Total) (Ci/yr)	Unabated Dose Rate (105-DR) [d] (mrem/yr)	Unabated Dose Rate (105-F)[d] (mrem/yr)
Am-241	1.24E-06	1.18E-09	6.20E-08	6.32E-08	8.26E-07	1.60E-06
Co-60	1.73E-06	1.64E-09	8.65E-08	8.81E-08	1.31E-08	2.51E-08
Cs-137	2.79E-04	2.65E-07	1.40E-05	1.42E-05	2.18E-06	4.19E-06
Eu-152	3.15E-06	2.99E-09	1.58E-07	1.60E-07	2.28E-08	4.36E-08
Eu-154	7.87E-07	7.48E-10	3.94E-08	4.01E-08	4.60E-09	8.81E-09
Sr-90	1.84E-04	1.75E-07	9.20E-06	9.37E-06	9.38E-07	1.80E-06
Pu-239/240	4.98E-06	4.73E-09	2.49E-07	2.54E-07	2.14E-06	4.14E-06
Pu-241	1.24E-05	1.18E-08	6.20E-07	6.32E-07	8.43E-08	1.63E-07

[a] Radionuclide annual possession quantities are as presented in BHI calculation 0100X-CA-V0020 (BHI 1998b).
[b] This represents 95% of the total annual possession quantity and uses a release fraction of 1×10^{-3}.
[c] This represents 5% of the total annual possession quantity and uses a release fraction of 1.
[d] The annual unabated dose was determined using the CAP-88 model. The PTE (Ci/yr) was the input from the model, and the model generated the annual unabated dose. The distance to MEI is 15,738 m (51,600 ft) to the east for the 105-DR Facility and 9,561 m (31,400 ft) to the east for the 105-F Facility. The CAP-88 model summary and synopsis are presented in BHI calculation 0100X-CA-V0020 (BHI 1998b).

The CAP-88 model was used to determine the annual unabated offsite dose. The PTE (Ci/yr) was the input for the computer model, and the model generated the annual unabated dose. The distance to the maximally exposed individual (MEI) used in the model was 15,738 m (51,600 ft) to the east for the 105-DR Reactor building and 9,561 m (31,400 ft) to the east for the 105-F Reactor building. The CAP-88 model summary and synopsis are presented in BHI calculation 0100X-CA-V0020 (BHI 1998b). The total unabated offsite dose from the D&D activities at the 105-DR Reactor building is 6.22E-06 mrem/yr and for the 105-F Reactor building is 1.20E-05 mrem/yr.

After the characterization activities (described in Section 4.3 of this document) are complete, the results will be evaluated to determine that the D&D activities are still within the assumptions made for the PTE calculations. If changes have occurred, this monitoring plan will be modified as necessary.

B.2.2 116-D, 116-DR, and 119-DR and Ducting

Radiological inventory data were acquired from the following UNC Nuclear Industries documents:

- UNI-3492, *ARCL Calculations for Decommissioning the 116-F Stack* (UNI 1985)
- UNI-3826, *ARCL Calculations for Decommissioning the 116-C Stack* (UNI 1986)
- UNI-3827, *ARCL Calculations for Decommissioning the 116-H Stack* (UNI 1987a).

These documents describe in situ contamination levels after the 116-C, 116-F, and 116–H stacks were explosively demolished and left in place in 1983. Samples were obtained at various heights from the interior of the stack, directly opposite of the plenum feed location. Most of the contamination was found directly opposite of the plenum feed location and in the areas just above this location. Contamination levels, as expected, decrease at higher elevations in the stack. The following assumptions were used to estimate a conservative radiological inventory for the 116-D and 116-DR stacks:

- The highest radioactive concentrations for each isotope were determined from the UNI reports (UNI 1985, 1986, and 1987a).

- The maximum inner surface area of the three stacks (116-C) was used. The 116-C stack inner surface is greater than or equal to the 116-D and 1106-DR stacks.

- Even though sampling showed that contamination was found only in the top 1 to 2 mm (0.04 to 0.08 in.) of the inside of the stacks, volumetric contamination was conservatively calculated for a depth of 0.635 cm (0.25 in.) throughout the entire stack.

The 119-DR Exhaust Air Sampling Building is relatively free of contamination. As stated in the RSR 105DR-99-0633, no dose rates within the building were above 0.5 mrem/hr (beta/gamma), all alpha surveys were below 100 dpm/100 cm^2, and all beta/gamma surveys were below 5,000 dpm/100 cm^2. To conservatively calculate a radiological inventory, the inside area of the small building was taken (i.e., four walls, floor, and a roof) and multiplied by 5,100 dpm/ 100 cm^2. The same isotopic abundance identified above for the stacks was assumed for 119-DR. The dimensions of the "Butler" building are approximately 4.2 m by 6 m by 2.4 m (14 ft by 20 ft by 8 ft) high. To take into account the slightly pitched roof, a height of 2.74 m (9 ft) was assumed. Details of this calculation are documented in BHI calculation 0100D-CA-V0064, *Air Emission Calculations for the 116-D & 116-DR Exhaust Stacks, 119-DR Sample Building and Associated Above-grade Ducts* (BHI 1999b).

The above-grade ducting contains low levels of contamination. The highest reading at the 116-DR and 116-D and associated above-grade ducting was 7,800 dpm/100 cm^2 (beta/gamma), which was found at 116-D. At 116-DR, the highest reading inside of the stack and above-grade ducting was 7,200 dpm/100 cm^2 (beta/gamma). To calculate a total activity, the inside area of the interior ducting was taken for both 116-D and 116-DR above-grade ducts and multiplied by 8,000 dpm/100 cm^2 (even though the maximum reading was 7,800 dpm/100 cm^2). Because the maximum radiological survey reading will be assumed over the entire inner surface area, this is considered conservative. With the exception of one isolated, low-activity area at both sites (116-D and 116-DR), no alpha contamination was found. The isotopic distribution calculated for the 116-D and 116-DR stacks will be used, which includes some alpha emitters, but this is yet another conservative assumption. Generally, for a respirable dose, alpha contamination isotopes contribute more dose per unit activity than beta or gamma radiation emitters. Details are documented in BHI calculation 0100D-CA-V0064 (BHI 1999b).

The radionuclide annual possession quantities and subsequent potential emission calculations are presented in Table B-2 for the D&D of the 116-D and 116-DR stacks, 119-DR Building, and associated above-grade ducting. It is assumed that 75% of the total annual possession quantity will represent the emissions from nonaggressive methods, and the release fraction of 1E-03 will be applied in accordance with WAC 246-247-030(21)(a). Twenty-five percent of the total annual possession quantity will represent the emissions from aggressive methods, and the release fraction of 1 will be applied per direction provided by the Washington State Department of Health (Conklin 1994). Details are documented in BHI calculation 0100D-CA-V0064 (BHI 1999b).

Table B-2. 116-D and 116-DR Stacks, 119-DR Building, and Ducting PTE Values.

Radionuclide	Annual Possession Quantity (Total) (Ci/yr)	PTE[d] (1E-3 RF) (Ci/yr)	PTE[e] (1RF) (Ci/yr)	PTE (Total) (Ci/yr)	Unabated Dose Rate (mrem/yr)
H-3 [b]	7.87E-04	5.90E-07	1.97E-04	1.97E-04	7.82E-09
C-14[a]	5.14E-03	3.86E-06	1.29E-03	1.29E-03	2.84E-06
Co-60[b]	2.51E-04	1.88E-07	6.27E-05	6.29E-05	9.33E-06
Sr-90[c]	3.72E-04	2.79E-07	9.30E-05	9.33E-05	9.34E-06
Cs-137 [b]	8.11E-03	6.08E-06	2.03E-03	2.03E-03	7.21E-05
Ba-137M	7.68E-03	5.76E-06	1.92E-03	1.92E-03	2.40E-04
Eu-152 [c]	7.46E-05	5.60E-08	1.87E-05	1.87E-05	2.66E-06
Pu-239 [c]	1.86E-05	1.40E-08	4.66E-06	4.67E-06	3.96E-05

Am-241 [b]	1.14E-04	8.54E-08	2.85E-05	2.86E-05	3.73E-04
--	2.25E-02	--	--	--	7.49E-04

[a] The maximum concentration for this nuclide was found at 116-H.
[b] The maximum concentration for this nuclide was found at 116-C.
[c] The maximum concentration for this nuclide was found at 116-F.
[d] Reference WAC 246-247-030 *for release fraction of* 1×10^{-3} *for nonaggressive activities.*
[e] Reference Washington State Department of Health's Air-94-802 for release fraction of 1 for aggressive activities.

The CAP-88 model was used to determine the annual unabated offsite dose. The PTE (Ci/yr) was the input for the computer model, and the model generated the annual unabated dose. The distance to the MEI used in the model was 15,738 m (51,600 ft) to the east. The CAP-88 model summary and synopsis are presented in BHI calculation 0100D-CA-V0064 (BHI 1999b). The total unabated offsite dose from the D&D activities at the 116-D and 116-DR stacks, 119-DR Building, and associated above-grade ducts is 7.49E-04 mrem/yr.

After the characterization activities (described in Section 4.3 of this document) are complete, the results will be evaluated to determine that the D&D activities are still within the assumptions made for the PTE calculations. If changes have occurred, this monitoring plan will be modified as necessary.

B.2.3 105-F Fuel Storage Basin

The radiological inventory for the FSB was acquired from BHI (1998c) (Table 4-3 for the sediment and Table 4-4 for the irradiated fuel elements). The inventories reflect data as of March 1, 1998. As stated in BHI (1998c), the radionuclide inventory in the water at the bottom of the FSB is negligible compared to the sediment; therefore, it was not included in this calculation. A relatively insignificant radionuclide inventory is assumed to be present within the FSB concrete walls, columns, and floor and is included in the inventory of the sediment, as stated in UNI (1987b). Some of the FSB inventory may have contaminated the soil underlying and adjacent to the FSB; therefore, minimal amounts of contaminants are expected to be present and are included as the sediment inventory. Therefore, the demolition of the FSB walls, columns, floor, and the soils underlying and adjacent to the FSB are included as part the FSB activities. If extensive soil contamination is found in adjacent and underlying soils, work will be deferred to the Remedial Action/Waste Disposal Project, with approval from the lead regulatory agency. As discussed in the 105-F FSB calculation (BHI 2001~~0~~), the FSB activities will occur over the period of 1 year.

The cesium-137 daughter product (i.e., barium-137m) and uranium-238 daughter products were calculated by the CAP88-PC code. The total PTE value for strontium-90 was also used for the total PTE value for yttrium-90 (which is in secular equilibrium and a daughter product of strontium-90).

Consistent with the methodology in BHI (1998c), the sediment inventory was multiplied by a factor of 1.75 for conservatism. It is estimated that 50,000 kg of sediment are present in approximately the bottom 0.9 m (3 ft) of the basin (BHI 1998c). The sediment exists in the bottom 0.9 m (3 ft) of the FSB at variable depths, and no radionuclide inventory has been shown to exist in the soil overburden (top 5.2 m [17 ft]). Basin well water measurements at approximately 0.9 m (3 ft) from the bottom of the basin showed 35% to 70% moisture above interstitial saturation conditions.

The release fraction (RF) for the sediment was obtained from the U.S. Environmental Protection Agency's Clearinghouse for Inventories and Emission Factors (CHIEF), *Compilation of Air Emission Factors,* AP-42, Table 11.9-4 (BHI 2001~~0~~). An RF value of 1.8×10^{-5} for truck loading of soil overburden by use of a power shovel (a large excavator) was used. This value is appropriate and bounding for the sediment, which is composed of fine sand, clay sediment, and iron oxides, and would undergo the same type of activities (e.g., scooping, loading, and handling material).

Table 4-4 of BHI (1998c) shows the inventory of five irradiated fuel elements. During deactivation and cleanout of the 105-D and 105-DR basins, a total of ~~five~~ 5 fuel elements were found; therefore, it was conservatively assumed in BHI (2001~~1998c~~) that 20~~twentyfive~~ irradiated fuel elements would be used for the inventory in the 105-F FSB. An RF of 1×10^{-6} was used for the fuel element for 80% of the inventory. As discussed in Section 4.2.1.3.4 of BHI (1998c), a fuel element may have a thin coating of uranium oxide present from corrosion that would be available for release. An RF of 1×10^{-3} will be used for 20% of the inventory for the thin coating (in a particulate form) that may be present on the outside of the fuel. These two RFs are consistent with the definitions found in WAC 246-247-030.

The total PTE was calculated by the sum of the PTE (sediment) and PTE (fuel) for each isotope and was used as the input to the CAP88-PC code. CAP88-PC, Version ~~1~~2.0, is used to calculate the dose to the MEI. The MEI was assumed to be located at the Hanford Site boundary at a compass bearing from the source that yielded the highest dose from all pathways, as computed by the CAP88-PC program. As shown in BHI (2001~~0~~), the distances to the Laser Interferometer Gravitational Wave Observatory (LIGO) and the Kaiser Aluminum Plant (313 Building) were considered when calculating the distances to the MEI. These two buildings were considered because they allow unrestricted public access. The LIGO, which is 18,701 m (61,358 ft) south of the 105-F Building, was used as the distance to the MEI versus 32,262 m (105,851.6 ft) to the south, which is the distance to the Hanford Site boundary. The Kaiser Aluminum Plant, which is 34,094 m (111,862.4 ft) to the south-southeast of the 105-F Building, was used as the distance to the MEI versus 36,510 m (119,789.3 ft) to the south-southeast, which is the distance to the Hanford Site boundary. The CAP88-PC model assumes exposure is on a 24-hour/day, 365-days/yr basis. Distances to the Site boundary were computed using the Hanford Geographic Information System. A Hanford Site-specific wind file was used to model releases in the 100 Areas. This wind file was based on data collected in the 100-F Area between 1986 and 1996 at the 10-m (32.8-ft) level. The FSB activities will occur over the period of 1 year.

The distance to the MEI used in the model was 10,314 m (33,840.2 ft) to the east-southeast of the 105-F Reactor. The CAP88-PC synopsis and summary are included in BHI (2001~~0~~). The total unabated offsite dose from 105-F FSB activities was estimated as ~~9.79×10^{-3}~~3.40×10^{-2} mrem/yr, as shown in Table B-3.

B.2.3.1 Total Combined Dose. The total combined dose for all planned activities in the 100-D and 100-DR Areas is 7.55E-04 mrem/yr ($6.22 \times 10^{-6} + 7.49 \times 10^{-4}$ mrem/yr). The total combined dose for all planned activities in the 100-F Area is ~~9.80×10^{-3}~~3.40×10^{-2} mrem/yr (1.20×10^{-5} + ~~9.79×10^{-3}~~3.40×10^{-2} mrem/yr).

B.2.4 Emission Controls

The D&D will consist of nonaggressive methods such as wiping a surface or applying foam polymers, or aggressive methods such as scabbling, abrasive blasting, vacuuming, and the use of portable ventilation filter units. For the aggressive methods, the equipment used will contain charcoal and/or HEPA-filtered units. By connecting a HEPA-filtered unit to the tools, the dust and debris can be collected into containers as it is generated. The HEPA-filtered units will have a minimum efficiency of 99.95% for the removal of airborne particulates. The basis for this is *The Nuclear Air Cleaning Handbook,* Section 8.2 (Burchsted et al. 1976).

The SSE will be a deactivated facility that is uninhabited and locked except during S&M activities, which are expected to occur on a 5-year basis. Many of the reactors' components will be removed as part of the stabilization effort for placing the facility into ISS. The reactor block's penetrations will be sealed during the ISS project. Many accessible areas of the SSE will have loose contamination removed and a fixative applied to limit the spread of contamination.

No forced ventilation of the building is necessary either during the ISS period or during periodic surveillance. A provision will be made to provide forced ventilation to the facility, if required, for maintenance. If forced ventilation is required, a portable skid-mounted exhauster will be used. This portable exhauster will either meet the conditions of the portable temporary radionuclide air emission unit Notice of Construction or a separate approval may be obtained prior to conducting the maintenance depending on current site conditions. The skid-mounted exhauster would be configured with HEPA filters.

B.2.5 Best Available Radionuclide Control Technology

The D&D activities have the potential to release radioactive emissions to the atmosphere. Implementing the BARCT for these radioactive emissions has been identified as an applicable or relevant and appropriate requirement.

For nonaggressive decontamination, the use of wiping or applying foam polymers is an ALARA control that has been accepted as BARCT for fugitive particulate radionuclide air emissions, particularly when the potential offsite dose is low. For aggressive techniques, HEPA filters are used to collect the dust generated. The use of HEPA filters has been generally accepted as BARCT. Because structure demolition may be a source of radioactive fugitive emissions, dust suppressants (e.g., water and fixatives) will be used and are considered BARCT for demolition. When using water, quantities used will be minimized to prevent water accumulation, puddles, and run-off within the area where the water is being used.

For the 105-F FSB, activities will primarily consist of heavy equipment excavating soil and sediment, demolishing the walls and columns, and possibly removing the floor. Fixatives (e.g., paint) may be applied to areas of contamination. Because excavation may be a source of radioactive fugitive emissions, dust suppressants (e.g., water and fixatives) will be used and are considered BARCT.

- Water will be applied during excavation, container loading, and backfilling processes to minimize airborne releases.

- Soil fixatives will be applied to any contaminated soils or soils that are being stockpiled (e.g., for reuse) that will be inactive for more than 24 hours.

- Fixatives will be applied to contaminated soils that will be inactive less than 24 hours at the end of work operations, if the sustained windspeed is predicted overnight to be 20 mph or greater, based on the Hanford Meteorological Station morning forecast. This will allow for enough time, if necessary, to prepare for the application of dust control measures. If a soil fixative has already been applied and the soil will remain undisturbed, further uses of fixatives will not be reapplied, unless needed. The fixatives or other controls will not be applied when the contaminated soils are frozen or it is raining, snowing, or other freezing precipitation is falling at the end of work operations.

- An entry will be made in the project logbook when the forecasts predict sustained wind speeds of 20 mph or greater, and dust control is to be applied at the end of the work shift.

B.2.6 Monitoring

The potential dose from D&D activities is not greater than 0.1 mrem/yr; therefore, this air emission source is not subject to the continuous monitoring requirements of the radionuclide National Emission Standards for Hazardous Air Pollutants. However, periodic confirmatory measurements will be taken throughout the duration of the project.

Periodic confirmatory measurements are defined as operating one low-volume air monitor on the upwind and downwind side of each reactor area. Placement of the monitors will be based on the predominant wind direction. The upwind and downwind monitors for the 105-DR Area are N492 and N493, respectively. The upwind and downwind monitors for the 105-F Area are N494 and N495, respectively. Figures B-1 and B-2 show the locations of the monitors at 100-DR and 100-F Areas.

The monitors will run continuously during D&D activities involving radiologically contaminated materials. The operation of these monitors will follow the protocol established for near-facility monitors on the Hanford Site. Basically, the samplers will run continuously for

2 weeks (or for the duration of the radiological work, if less than 2 weeks). The samples will then be changed and sent to the designated laboratory for total alpha and total beta analysis. The samples will be composited quarterly and analyzed for gamma energy analysis radionuclides, strontium-90, isotopic uranium, and isotopic plutonium. The data results will then be entered into the Automated Bar Coding of Air Samples at Hanford (ABCASH) database for record keeping and reporting. The data from these monitors will be included in the annual reports prepared for the Hanford Site.

During the demolition of the 116-D stack and associated above-grade ducting, an additional downwind air sampler will be operational during the duration of demolition activities. The existing upwind air monitor currently being used at 105-DR will be used for the 116-D stack and associated above-grade ducting D&D activities. The existing upwind and downwind monitors (see Figure B-2) currently being used for 105-DR ISS Project activities will be used to monitor D&D activities for the 116-DR, 119-DR, and associated ducting. The air monitors used for the 105-F ISS activities, as shown in Figure B-1, will also be used for 105-F FSB activities. Operation of the air samplers will follow the established protocol, as described previously.

Table B-3. 105-F Fuel Storage Basin PTE and Unabated Offsite Dose Values.[d]

Isotope	Sediment		Fuel				Sediment + Fuel = Total	Unabated Offsite Dose (mrem/yr)[c]
	APQ (Ci/year)	PTE (Sediment) (Ci/yr)[a]	APQ (Ci/year)	PTE (1E-3) (Ci/yr)[b]	PTE (1E-6) (Ci/yr)[b]	PTE (Fuel) (Ci/yr)	Total PTE (Ci/yr)	
Ni-59	8.75E-01	1.57E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.57E-05	6.61E-09
Co-60	3.48E+00	6.27E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.27E-05	1.41E-05
Ni-63	9.60E+01	1.73E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.73E-03	8.00E-07
Sr-90	1.79E+01	3.22E-04	~~2.93E+01~~ 1.17E+02	~~5.86E-03~~ 2.34E-02	~~2.34E-05~~ 9.38E-05	~~5.88E-03~~ 2.35E-02	~~6.21E-03~~ 2.39E-02	~~9.44E-04~~ 3.63E-03
Y-90[e]	--	--	--	--	--	--	~~6.21E-03~~ 2.39E-02	~~2.08E-06~~ 8.11E-06
Cs-137	2.08E+01	3.75E-04	~~2.97E+01~~ 1.19E+02	~~5.94E-03~~ 2.38E-02	~~2.38E-05~~ 9.50E-05	~~5.96E-03~~ 2.39E-02	~~6.34E-03~~ 2.42E-02	~~3.42E-04~~ 1.30E-03
Ba-137[f]	--	--	--	--	--	--	--	~~1.14E-03~~ 4.35E-03
Eu-152	1.47E+01	2.65E-04	~~1.52E-04~~ 6.08E-04	~~3.04E-08~~ 1.22E-07	~~1.22E-10~~ 4.86E-10	~~3.05E-08~~ 1.22E-07	2.65E-04	5.72E-05
Eu-154	2.54E+00	4.57E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.57E-05	7.97E-06
U-238[g]	1.58E-02	2.84E-07	~~6.00E-03~~ 2.40E-02	~~1.20E-06~~ 4.80E-06	~~4.80E-09~~ 1.92E-08	~~1.20E-06~~ 4.82E-06	~~1.49E-06~~ 5.10E-06	~~6.51E-06~~ 2.23E-05
Pu-239	4.55E+00	8.19E-05	~~1.20E+00~~ 4.80E+00	~~2.40E-04~~ 9.60E-04	~~9.60E-07~~ 3.84E-06	~~2.41E-04~~ 9.64E-04	~~3.23E-04~~ 1.05E-03	~~4.20E-03~~ 1.37E-02
Am-241	8.58E-01	1.54E-05	~~3.92E-01~~ 1.57E+00	~~7.84E-05~~ 3.14E-04	~~3.14E-07~~ 1.25E-06	~~7.87E-05~~ 3.15E-04	~~9.41E-05~~ 3.30E-04	~~1.88E-03~~ 6.61E-03
Pu-238	4.66E-01	8.40E-06	~~1.80E-02~~ 7.20E-02	~~3.60E-06~~ 1.44E-05	~~1.44E-08~~ 5.76E-08	~~3.61E-06~~ 1.45E-05	~~1.20E-05~~ 2.29E-05	~~1.45E-04~~ 2.76E-04
Pu-241	9.87E+00	1.78E-04	~~5.34E+00~~ 2.14E+01	~~1.07E-03~~ 4.27E-03	~~4.27E-06~~ 1.71E-05	~~1.07E-03~~ 4.29E-03	~~1.25E-03~~ 4.47E-03	~~2.55E-04~~ 9.12E-04
Pu-240	0.00E+00	0.00E+00	~~3.00E-01~~ 1.20E+00	~~6.00E-05~~ 2.40E-04	~~2.40E-07~~ 9.60E-07	~~6.02E-05~~ 2.41E-04	~~6.02E-05~~ 2.41E-04	~~7.83E-04~~ 3.13E-03
Kr-85	0.00E+00	0.00E+00	~~8.61E-01~~ 3.44E+00	~~1.72E-04~~ 6.89E-04	~~6.89E-07~~ 2.76E-06	~~1.73E-04~~ 6.92E-04	~~1.73E-04~~ 6.92E-04	~~2.51E-11~~ 1.00E-10
Sm-151	0.00E+00	0.00E+00	~~3.61E-01~~ 1.44E+00	~~7.22E-05~~ 2.89E-04	~~2.89E-07~~ 1.16E-06	~~7.25E-05~~ 2.90E-04	~~7.25E-05~~ 2.90E-04	~~8.77E-08~~ 3.51E-07
Cd-113	0.00E+00	0.00E+00	~~2.00E-03~~ 8.00E-03	~~4.00E-07~~ 1.60E-06	~~1.60E-09~~ 6.40E-09	~~4.02E-07~~ 1.61E-06	~~4.02E-07~~ 1.61E-06	0

Table B-3. 105-F Fuel Storage Basin PTE and Unabated Offsite Dose Values.[d]

Isotope	Sediment		Fuel				Sediment + Fuel = Total	Unabated Offsite Dose (mrem/yr)[c]
	APQ (Ci/year)	PTE (Sediment) (Ci/yr)[a]	APQ (Ci/year)	PTE (1E-3) (Ci/yr)[b]	PTE (1E-6) (Ci/yr)[b]	PTE (Fuel) (Ci/yr)	Total PTE (Ci/yr)	
Nb-94	0.00E+00	0.00E+00	~~8.00E-04~~ 3.20E-03	~~1.60E-07~~ 6.40E-07	~~6.40E-10~~ 2.56E-09	~~1.61E-07~~ 6.43E-07	~~1.61E-07~~ 6.43E-07	~~1.57E-07~~ 6.25E-07
Se-79	0.00E+00	0.00E+00	~~2.00E-04~~ 8.00E-04	~~4.00E-08~~ 1.60E-07	~~1.60E-10~~ 6.40E-10	~~4.02E-08~~ 1.61E-07	~~4.02E-08~~ 1.61E-07	0
Pd-107	0.00E+00	0.00E+00	~~2.00E-05~~ 8.00E-05	~~4.00E-09~~ 1.60E-08	~~1.60E-11~~ 6.40E-11	~~4.02E-09~~ 1.61E-08	~~4.02E-09~~ 1.61E-08	~~2.40E-12~~ 9.63E-12
Tc-99	0.00E+00	0.00E+00	~~1.00E+00~~ 4.00E+00	~~2.00E-04~~ 8.00E-04	~~8.00E-07~~ 3.20E-06	~~2.01E-04~~ 8.03E-04	~~2.01E-04~~ 8.03E-04	~~6.42E-06~~ 2.56E-05
Zr-93	0.00E+00	0.00E+00	~~2.00E-03~~ 8.00E-03	~~4.00E-07~~ 1.60E-06	~~1.60E-09~~ 6.40E-09	~~4.02E-07~~ 1.61E-06	~~4.02E-07~~ 1.61E-06	~~8.48E-10~~ 3.40E-09
							Total	~~9.79E-03~~ 3.40E-02

[a] The sediment is assumed to have a release fraction of 1.8×10^{-5}, as stated in AP-42, Table 11.9-4.
[b] Conservatively, 80% of the fuel is assumed to have a release fraction of 1×10^{-6} and 20% of the fuel is assumed to have a release fraction of 1×10^{-3}.
[c] The annual unabated dose was determined using the CAP-88 model. The PTE (Ci/yr) was the input and the model generated the annual unabated dose. The distance to the MEI is 10,314 m (33,840.2 ft) to the east-southeast for the 105-F Facility. The CAP-88 model summary and synopsis is presented in BHI (2001~~0~~).
[d] The information in this table was obtained from BHI (2001~~0a~~).
[e] As discussed in Section B.2.0, the total PTE for Sr-90 was used for Y-90.
[f] As discussed in Section B.2.0, the Cs-137 daughter product (Ba-137) was calculated by the CAP-88PC code.
[g] As discussed in Section B.2.0, daughter products of U-238 were calculated by the CAP88-PC code.
APQ = annual possession quantity
RF = release fraction

Figure B-1. 105-F Air Monitoring.

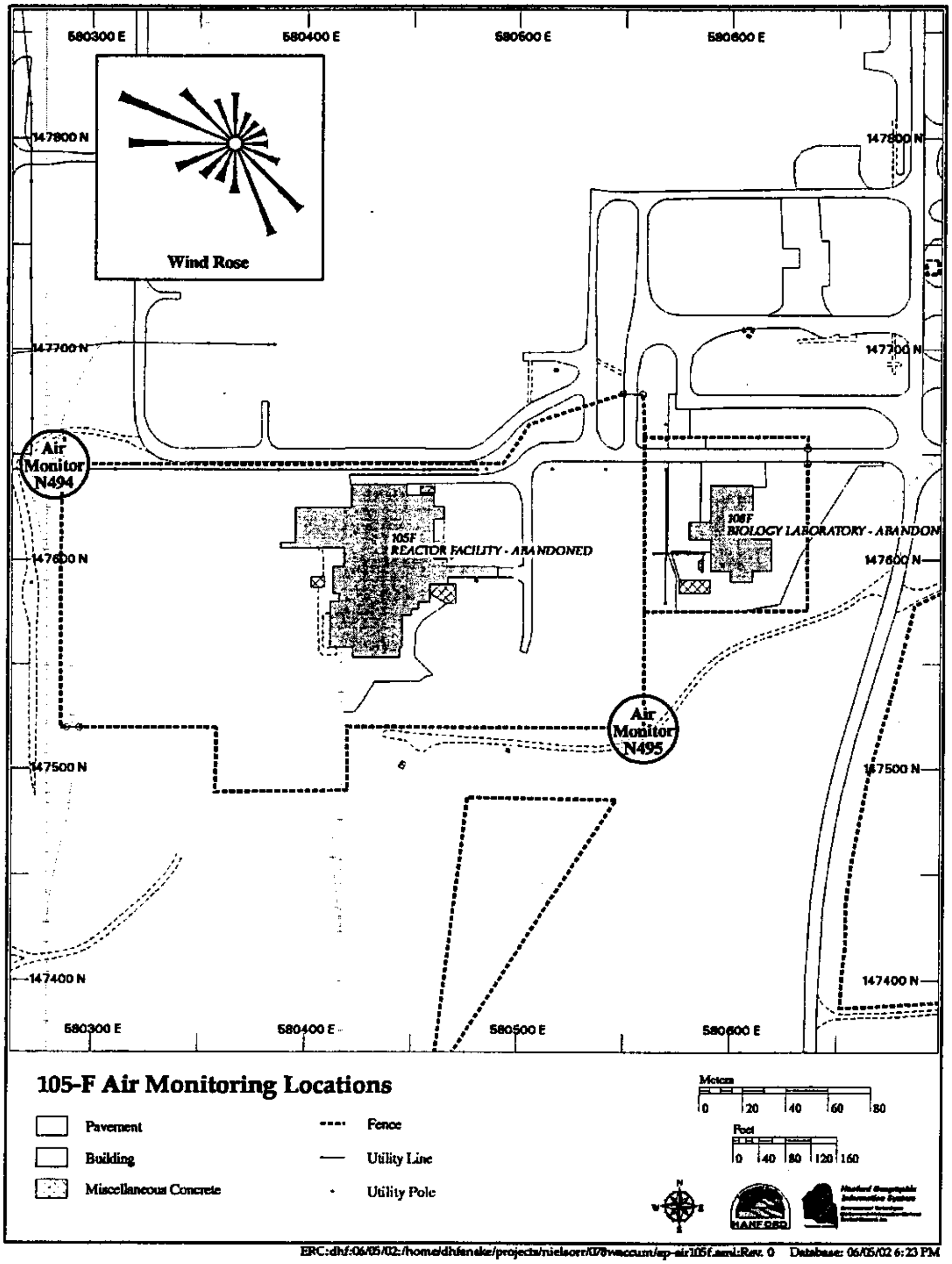
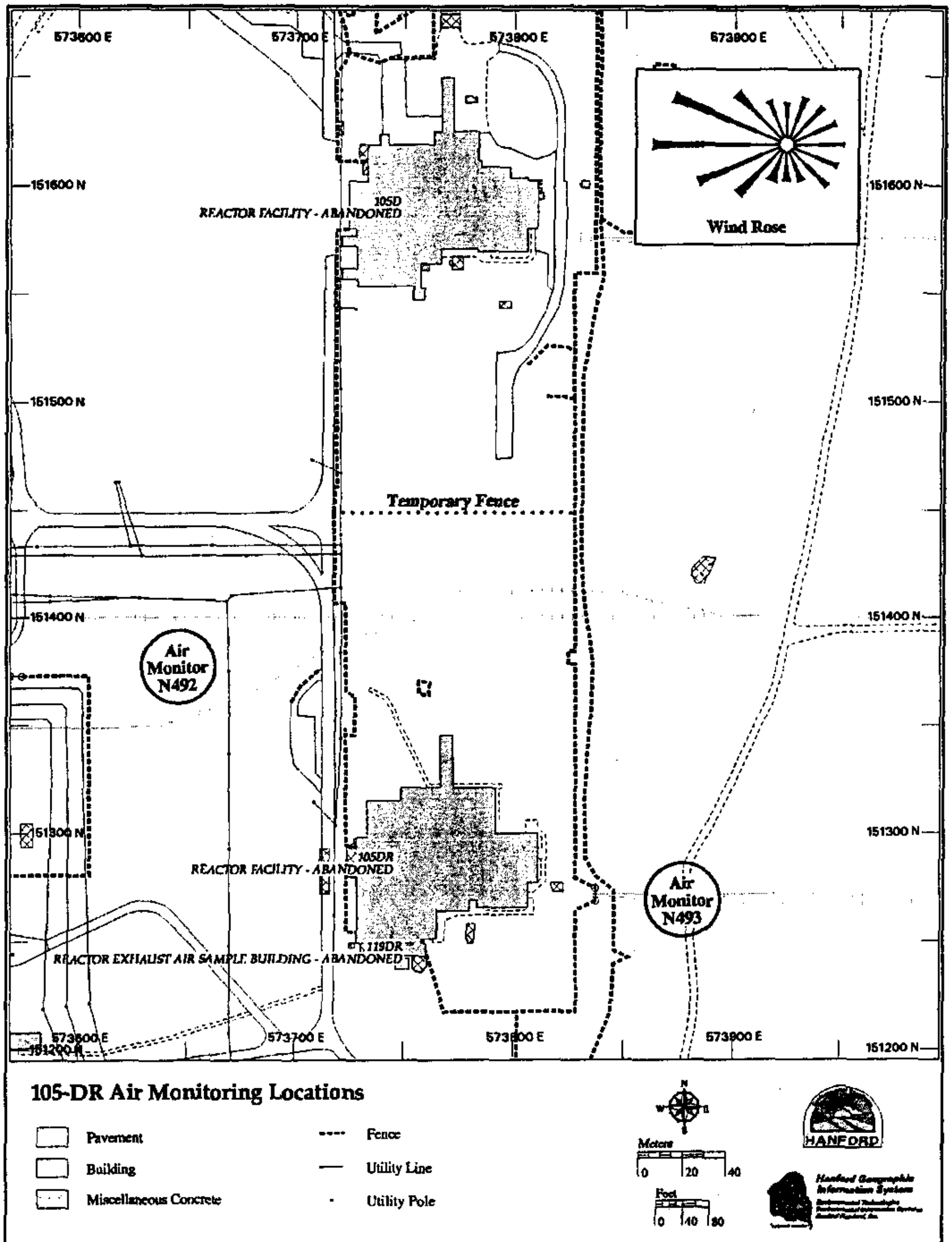


Figure B-2. ~~105-D and~~ 105-DR Air Monitoring.



ERC:dhf:06/10/02:/home/dhfenske/projects/nielsen/078waccum/ap-air105dr.aml:Rev. 0 Database: 06/10/02 10:29 AM

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APPENDIX C

105-F FUEL STORAGE BASIN DECOMMISSIONING TECHNICAL APPROACH

APPENDIX C

105-F REACTOR FUEL STORAGE BASIN DECOMMISSIONING TECHNICAL APPROACH

C.1 BACKGROUND

The 105-F Reactor fuel storage basin (FSB) is a large, reinforced-concrete basin that served as a collection, storage, and transfer facility for the irradiated fuel elements discharged from the reactor. The FSB contained water that acted as a coolant and provided shielding of irradiated fuel.

The FSB is roughly 24 m by 24 m by 6.1 m (80-ft by 80 ft by 20 ft) deep, with a transfer pit that is approximately 7.6 m (25 ft) deep and approximately 0.79 m^2 (8.5 ft^2). The fuel was stored in aluminum or stainless steel baskets, approximately 46.7 cm by 55.9 cm by 52.1 cm (18.4 in. by 22 in. by 20.5 in.) high. The baskets were held in rows each separated by concrete walls, 86.4 cm (34 in.) high by 35.6 cm (14 in.) thick and approximately 86.4 cm (34 in.) apart, running the length of the basin. The floor of the basin is nominally 30.5- to 61-cm (12- to 24-in.)-thick concrete.

Following shutdown and in preparation for decommissioning in 1970, the water level was reduced to approximately 0.6 to 0.9 m (2 to 3 ft) deep and some of the fuel storage baskets were removed. Sediment buildup from 20 years of operations remains on the bottom of the basin. The decking, 5.1-cm by 20.3-cm (2-in. by 8-in.) wood planks assembled in 0.9-m by 3.7-m (3-ft by 12-ft) sections that formed the walking deck above the basin, was dropped into the basin. Access to the fuel baskets was originally provided by a monorail system above each aisle that penetrated the decking. The monorails, steel "T" sections 9.05 cm (3.56 in.) wide by 6.51 cm (2.56 in.) high, were cut from the ceiling and also dropped into the basin.

Other miscellaneous items were also placed in the FSB with a sense of finality; thus, little effort was made to record the items placed in the basin, and no attempt was made to place the items in an orderly manner. The FSB was then filled to the top with approximately 3,442.5 m^3 (4,500 yd^3) of local surface material (essentially clean, fine sand). Although an inventory was taken to list some materials that were left in the basin, complete fuel elements or fragmented pieces that could remain cannot be ruled out based on the experience at other FSBs at the other Hanford Site production reactors.

The debris that remains in the basin (e.g., monorails, decking, fuel buckets, process tubes, and possibly fuel elements and pieces) are expected to be found primarily in the bottom 0.9 m (3 ft) of the basin. These items were left in or dropped into the basin prior to adding the fill to the basin. While some decking or monorail material may be encountered above the 0.9-m (3-ft) level, history and some preliminary characterization done in about 1990 indicate that the top 5.2 m (17 ft) of fill should be expected to be free of radiological or chemical contamination.

The cleanout of the FSB (as part of the facility decommissioning) is being planned in two stages. Stage I will address the FSB from the surface to 5.2 m (17 ft) below grade. Stage II will address the remaining 0.9 m (3 ft). A description of the proposed technical approach for the FSB cleanout follows.

C.2 ENGINEERING AND PLANNING

Some of these actions began in October 1999; however, all of these actions are needed to complete the definitive design and preparation to allow the start of physical demolition/cleanout work in late fiscal year 2000. The scope of the engineering/planning work consists of the following items:

1. Review of the auditable safety analysis to ensure that the methods and plans being proposed are either covered in the current analysis (expected case) or by a newly generated Management of Change review.

2. Revise the removal action work plan and air monitoring plan to reflect planned actions for Stages I and II. Submit to the U.S. Department of Energy and regulators to allow sufficient time for approval prior to the start of physical work in September 2000.

3. Define the water and waste management strategies for Stages I and II. This will include incorporating the appropriate scope in other project scopes for interface items and will provide the basis for field implementation documents.

4. Prepare the data quality objective summary reports and the sampling and analysis plans for Stages I and II fill and debris removal. Stage II documents will also address remaining basin concrete and underlying soils.

5. Perform Stage I sampling of the top 5.2 m (17 ft) of the basin fill.

6. Coordinate with the Project Hanford Management Contractor to ensure that any fuel found is properly identified, controlled, packaged, and transported for inclusion with other spent fuels program inventory.

7. Develop purchase order packages for the Brokk® 330 and In Situ Object Counting System (ISOCS) units.

8. Procure and test the Brokk 330.

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9. Prepare field procedures for the Brokk 330, ISOCS, 3-D GammaCam™, Laser-Assisted Ranging and Data System (LARADS), and other instruments for use in the FSB.

10. Fabricate, install, and operate the water-removal system within the FSB. Remove, test, and dispose of up to 45,600 L (12,000 gal) of contaminated water at the 200 Area Effluent Treatment Facility.

11. Specify and procure waste containers and casks to support Stage II fuel, debris, and fill removal.

C.3 UPPER FILL REMOVAL (STAGE I)

The initial steps of the cleanout address removal of the above-grade structure to facilitate access to the FSB. The removal of the FSB upper fill from grade down to -52.m (-17 ft) (i.e., the approximate height of the 86.4-cm [34-in.] separation wall that separated rows of fuel baskets) is also included. Elements of work include the following:

1. Demolish above-grade structure to allow access to the upper fill overburden.

2. Procure and perform acceptance testing on the ISOCS unit.

3. Excavate backfill from grade to -5.2 m (-17 ft) (approximately) with excavator.

4. Stockpile the fill from grade-level to approximately -4.6 m (-15 ft). Sample the stockpile in accordance with the Phase 4 Stage I sampling and analysis plan to verify acceptability for reuse.

Some materials and debris from this initial work will be loaded into containers and disposed at the Environmental Restoration Disposal Facility (ERDF), but attempts are planned to verify the stockpiled fill material meets standards for reuse. Stage I is scheduled to be completed by April 2001.

C.4 DEBRIS AND CONTAMINATED FSB FILL REMOVAL (STAGE II)

The next stage of the cleanout (to be completed by October 2001) addresses removal of the remaining contents (i.e., backfill and debris) from the 86.4-cm (34-in.) separation walls to the FSB floor. This includes the deployment of the Brokk 330 unit, 3-D GammaCam, ISOCS, LARADS, and/or other instruments . Elements of work include the following:

1. Remove protruding debris (e.g., columns and planking) using the Brokk 330 or an excavator with a shear, grapple, and/or a bucket with thumb attachment to achieve a relatively level working surface.

2. Characterize the remaining materials with the GammCam, ISOCS, LARADS, or other instruments. The purpose of this characterization will be to map the high-dose locations and determine the hot spots that may contain irradiated material (fuel and nonfuel) with the GammaCam or LARADS. The hot spots may then be concentrated on using the ISOCS or other equipment to determine if the hot spot meets a pre-determined fingerprint for fuel.

3. Retrieve any expected fuel elements or hot spot materials using the Brokk 330 and/or a large excavator with attachments, or personnel with long-handled tools. Fuel will be excavated, staged in a holding container, processed, and loaded into shiping containers for removal. Other hot spot materials will be excavated and packaged for disposal.

   Identifying, locating, and removing fuel elements and other hot spot materials are a first priority and are closely controlled by limitations in the project authorization (safety) basis documents.

4. Perform Stage II sampling of the bottom 0.9 m (3 ft) of sludge and fill material for waste designation purposes.

5. Remove the remaining backfill, debris, and sludge using the Brokk 330 with various attachments (e.g., bucket and grapple) or other equipment as appropriate (e.g., large excavator). In-process screening will be conducted in areas where shielding material is excavated that could have hidden high-dose or fuel pieces. These items will be packaged appropriately for disposal.

6. Package the materials (except fuel and transuranic waste), as appropriate, for disposal at the ERDF. All debris, contaminated materials and fill from the bottom 86.4 cm (34 in.) of the basin (except fuel and transuranic waste) will be appropriately processed and disposed at the ERDF. Additional processing may include macro-encapsulation of lead and/or high-dose materials (e.g., sludge). Any transuranic waste encountered will be stored at the Central Waste Complex.

7. Demolish the FSB, perform Stage II sampling of surrounding soils, and perform RESidual RADioactivity dose model analysis to verify that the appropriate release criteria have been met.

8. Backfill the area to approximate final grade. Final grading and surfacing will be completed after the installation of the safe storage enclosure roof.

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